NUCLEAR SAFETY

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foreword

This new quarterly in the U. S. Atomic Energy Commission's Technical Progress Review Series has been prepared by the Oak Ridge National Laboratory at the request of the Commission's Division of Information Services. It is intended to assist those interested in keeping abreast of significant developments in the field of nuclear safety. *Nuclear Safety* is not intended to be a comprehensive abstract of all literature published in this field during a given quarter; rather it is a mechanism for presenting concise reviews of selected subjects from time to time as prevailing interest and available information warrant.

Coverage of the Review is limited to topics relevant to the analysis and control of hazards associated with nuclear reactors; operations involving fissionable materials; and the products of nuclear fission. Primary emphasis is on safety in reactor design, construction, and operation; however, safety considerations in reactor fuel fabrication, spent fuel processing, nuclear waste disposal, and related operations are also treated. Safety in the use of radioisotopes in industry, medicine, and research is excluded as are most topics considered the province of health physics. Even with these exclusions, nuclear safety cuts across such diverse fields as nuclear physics, solid state physics, mechanics, chemistry, meteorology, geology, seismology, metallurgy, law, and nearly all branches of engineering. The authors will therefore review material from these fields which, in their opinion, has a direct bearing on nuclear safety.

In planning this first issue the editors made two arbitrary decisions. The first of these was that no attempt would be made to bring the reader up to date regarding developments of the past five or ten years. The second, and this was largely a matter of administrative convenience, was that the Review would be nominally limited to literature received by the Oak Ridge National Laboratory during the first five months of 1959. Some references to earlier literature are cited, however, and the reader is also referred to the wealth of nuclear safety information presented at the 1958 Geneva Conference. Subsequent issues of *Nuclear Safety* will review material issued during the quarter preceding manuscript preparation.

Interpretations, where given, represent the opinions of the editors who are employed by the Oak Ridge National Laboratory. Readers are urged to consult the references to original work for more complete information. Many members of the ORNL staff wrote review material, reviewed manuscript, or otherwise contributed to this publication. Their contributions are gratefully acknowledged.

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Biological and Economic Considerations

There is a broad spectrum of opinion as to the proper degree of emphasis on different criteria for evaluating the hazards involved in utilizing nuclear energy. The individual approaches express individual judgments as to the "most proper," "most desirable," or "most practical" combinations of economic, biological, and philosophical considerations.

One approach is exemplified in a recent analysis by Siddall.[1] He suggests that, as a solution to the economic problem of the cost of protection, a quantitative balance be established between the probable consequences of accidents and the cost of safety measures. The balance would include both humanitarian aspects and purely economic factors. He outlines the present acceptance of hazards, accidents, and deaths in "conventional" industries and in other activities and analyzes the acceptance quantitatively on an economic basis. For example,[2] the general public in the United States has been accepting an accident death rate in private motoring of about one death per 30 million passenger miles (or one death per $930,000, at 3.5 cents per passenger-mile). Statistics are introduced to show that the "safety index" of all industries combined is about $50 million per accidental death, with individual cases ranging from $5.2 million for coal mining to $200 million for automobile manufacturing. The electrical industry, including the mining and transportation of coal, is pointed out as a particularly pertinent case, with an index of about $10 million per accidental death.

A target index of $50 million per death is proposed for the nuclear energy industry. A particular example would be a 200 Mw(e) nuclear electric power station operating at a 0.8 load factor and selling power at 6 mills/kw-hr. An average of 0.17 deaths per year would be accepted on the proposed basis ($8.4 million/ $50 million = 0.17).

On a purely economic basis, it is submitted that the minimum sum of the cost of accidents plus the cost of accident prevention is a logical criterion. The total cost of accidents includes loss of revenue during down-time of the disabled plant, costs of repairs and lost equipment, and costs of compensation for personal and property damage. Costs of accident prevention include money spent on research and development in accident prevention fields, on more expensive materials, larger safety factors, multiplication and diversification of safety-sensitive components, and on enclosing shells and exclusion areas.

Accident costs and the costs of accident prevention can be reduced to a common basis, such as equivalent annual charges by application of the correct interest and repayment rates. In aiming for the lowest total cost, it is submitted that money spent on reducing accidents is more useful than that spent on containing accidents, particularly if the container makes rescue and repair work more difficult, as may be the case.

The analysis proposes that 2 per cent of the gross revenue is a reasonable target cost of accidents, corresponding to an insurance premium. It is proposed that further safety measures are probably not justified once the accident cost is reduced to this level. On this basis, the target accident cost for a 200 Mw(e) nuclear plant would be $168,000 per year.

Referring to AEC report WASH-740[3] for data on consequences of major accidents, Siddall's analysis derives simultaneous economic and humanitarian (deaths per year) equations in terms of the preceding parameters to calculate

the particular case of the 200 Mw(e) plant cited above. The following criteria result.

1. The primary coolant system must be designed so that it will not burst more often than an average of once in 50 years.

2. The protective system must not completely fail more often than on one occasion in 500, despite failures of the operational controls and bursting of the primary coolant container; partial failures must not be experienced on more than one occasion in 50.

3. The operational control system must not fail at full speed in the direction of increased reactivity more often than once in 160 years. Partial failures of this type can be tolerated once in 16 years.

It is submitted that, in the light of present knowledge and techniques, these targets seem to be reasonably attainable.

The author notes that the probable errors in employing this process are large. He believes, however, that this defect would be offset by the warnings provided by minor incidents (a conclusion not fully shared by some, as indicated in Beck's comments). He proposes that in control systems, for example, statistical experience can be built up on minor mishaps that cause no significant damage.

The conclusion of the analysis is that: "Meeting of performance targets can be checked from experience in time to avoid major accidents. If targets are met, money spent on a containing shell and consequent additions would be wasted. Economically, the reduction in cost of accidents would be less than the annual charges on the shell. From the humanitarian viewpoint, the money would have been better spent on safety measures in a dangerous industry, or better still, on medical research."

Comments on Siddall's approach were made[4] by each of four men of international stature in the field of nuclear energy.* Each complimented Siddall's report[1] as a significant contribution to the philosophy of approach in this difficult subject of placing reasonable limitations on the costs of nuclear safety. Beck, while commending the bold approach, warned of some practical difficulties of attempting to use the proposals submitted.

1. Since the parameters cover such a wide range of values and the values are so uncertain, confidence in the calculated results would be low.

2. Some of the calculational manipulations and procedures are "dubious."

3. There is some question that one could safely depend on the occurrence of small and intermediate accidents to warn of impending major accidents.

4. Many of the people who would be affected by nuclear accidents would not have accepted the risks voluntarily, as does a coal miner or factory worker, for example.

5. The extent of genetic damage to significant portions of future generations has not been determined.

The importance of learning to live with the potentially hazardous technology of nuclear energy in order to enjoy its benefits has been discussed from the standpoint of the biologist by Bruner.[5] As he points out, the excellent safety record of the nuclear industry demonstrates that it is possible to develop and put into practice a complete health-protection program which does not impede operations but which could result in award-winning safety records anywhere in the world. The solution to the problem lies in the application and expansion of approaches and attitudes toward health protection developed in the early days of the Manhattan Project. These included:

1. A broadly conceived research program to obtain more information on the biological effects of ionizing radiation.

2. Active participation of medical, safety, and health-protection staffs in the planning and designing of processes and equipment.

3. Incorporation of recommended safety factors regardless of cost.

4. Continued detailed medical supervision of operations and personnel.

Although experience indicates that "radiation problems," if met intelligently and vigilantly,

*C. K. Beck, Chief, Hazards Evaluation Branch, U. S. Atomic Energy Commission; W. B. Lewis, Atomic Energy of Canada, Limited; Manson Benedict, Department of Nuclear Engineering, Massachusetts Institute of Technology, and member of the AEC Advisory Committee on Reactor Safeguards; and Gunnar Randers, Director, Joint Establishment for Nuclear Energy Research, Lillestrom, Norway.

need not stand in the way of the large-scale development of atomic energy, the records also indicate that foolish and unpredictable chance exposures probably never will be completely eliminated. Thus the expansion of the nuclear industry and the accompanying scatter of accidents will undoubtedly lead to an increase in human cost. The continuation of effective developmental and basic research at a rate proportional to the number of reactors and amounts of byproducts may well decrease the percentage incidence so as to hold the absolute number of accidents to a negligible level. As long as an approach is followed which places biological acceptability ahead of monetary and engineering considerations it will be possible to achieve and maintain control of radiation and radioactivity such that the ecological systems of the surfaces of the earth can remain undamaged.

It is seen from the foregoing that the approach to nuclear-safety planning recommended by a biologist places biological considerations above those of economics and engineering, which is in sharp contrast to the approach suggested by Siddall[1] of striking a quantitative balance between the cost of safety measures and the consequences of a nuclear accident. A third approach, differing from both of these, is suggested by Stubbart,[6] an engineer with Gibbs and Hill, Inc., who argues that one should include only those safety measures that the customers demand, or, in other words, "meet the competition." He uses as his argument the fact that European engineers and managers of European utilities have satisfied themselves that the gas-cooled reactor plant without a container type of building meets their safety requirements, and yet Americans are insisting that they accept more stringent measures. He concludes that we should give the Europeans what they want and adjust our designs to satisfy their requirements. If an enclosure is not required by the specifications, it should not be included in the vendor's bid. Conversely, if the site requires an enclosure it should be designed by the client's consulting engineer and priced with the structure and not with the reactor plant.

Such an approach tends to overlook the fact that different types of reactors may require different degrees of containment and that the safety standards set by European customers may not be applicable to American-built reactors. The problem of radiation protection transcends national boundaries since a major release of radioactivity from an American-built plant would be a serious setback to the United States nuclear industry whether the plant were built in Europe or the United States. Nuclear plants constructed in foreign countries, therefore, should be built to meet the same safety criteria as those established for domestic plants. It remains only to establish what these criteria should be.

Stubbart presents the following points to show that a pressurized-water reactor is inherently safer than a gas-cooled graphite reactor of the Calder Hall type.

1. For meltdown of fuel elements to occur in a pressurized-water reactor, the surface temperature of the stainless-steel or zirconium cladding must reach 2700 to 3300°F compared with about 1200°F for the magnesium-alloy-clad fuel of the type used in present gas-cooled reactors.

2. More attention to detail is usually given in the design of a high-pressure system than a low-pressure system in recognition of a hazardous situation. In any event, piping systems for both high and low pressures are designed with the same safety factors.

3. A gas-cooled reactor contains no liquid to absorb heat through evaporation, whereas water left in the reactor vessel after a rupture has heat-removal capability through evaporation.

4. A quick-acting chemical fluid neutron poison can be used in a pressurized-water reactor to provide an additional increment of safety. There are no practical poisoning systems for gas-cooled reactors.

These arguments in favor of a pressurized-water reactor are open to question. For example, first, point 1 depends on the choice of cladding rather than the type of reactor. Stainless-steel-clad fuel elements can be used in gas-cooled reactors to eliminate the difficulty introduced through the use of a low-melting cladding material. Second, the primary consideration in comparing high- and low-pressure systems is not the relative integrity of the two piping systems, but the fact that if a rupture did occur, vaporization of the high-pressure coolant would result in greater dispersion of

radioactive material. This would be especially true if the reactor were not completely contained.

Designers of gas-cooled reactors refute points 3 and 4 by pointing out that the low power density of gas-cooled reactors, though objectionable from an economic standpoint, results in greater safety since natural circulation of the gas is adequate to remove the heat generated after shutdown. For this reason, an emergency shutdown system employing a fluid poison, such as proposed for the pressurized-water system, is believed not to be necessary. (J. A. Lane)

Site-selection Criteria

Criteria for selecting a reactor site have never been specifically defined. In practice, site approval is in the hands of the AEC safeguard groups (see page 40) which judge each case on its own merits. Senator Clinton P. Anderson,[7] Chairman, Joint Committee on Atomic Energy, stated in March, "I believe we are about at the stage where the Congress should consider whether to require the AEC to establish (by means of a regulation) criteria for the selection of sites for atomic power plants. Along with the criteria might be a requirement that applicants for a construction permit must submit alternative site proposals, with a further proviso that no permit should initially be issued for a site tentatively disapproved by the Advisory Committee on Reactor Safeguards."

In May 1959, the AEC published, in the *Federal Register*,[8] a notice concerning reactor-site criteria. Factors considered in site evaluation for power and test reactors are the following: exclusion distance around power and test reactors, population density in surrounding areas, meteorology, seismology, hydrology and geology, and the interrelation of these factors. The entire notice is reprinted as an appendix at the end of this section.

This notice, however, contains only a general reference to the influence of the reactor design on the choice of a suitable site. This aspect of site selection has been treated in more detail by Beck.[9]

"Basically, the biggest variable in the choice of sites for reactors is the reactor itself. . . . The four major features of the reactor and its associated equipment which must be known for a given site to be evaluated are as follows:

1. The quantity, identity, and physical nature of the fission-product inventory.

2. The quantity and identity of the radioactivity releases which may result from routine operations.

3. The likelihood of accidental releases, the nature of the possible accidents, and the portion of the fission-product inventory that might be released.

4. The special containment, shielding, missile, and other protection features afforded by the facility. . . .

"The quantity of fission products contained in a reactor is directly related to the power level at which the reactor is operated. After short periods of operation, about 10 curies per thermal watt of power level is in the reactor at shutdown, but this builds up to only five or six times as much after extended operation at the same power level. On the other hand, the nature of the radioactivity inventory at a given power level is directly dependent on the irradiation time or on the length of the fuel cycle. After 1 hr of operation the radioactivity of the fission products is mostly short-lived; in one year it would decay by a factor of about one million. After one year of operation, the proportionate amount of long-lived fission products is much higher; decay for a year after such operation reduces the level by a factor of less than 100." (J. A. Lane)

The notice in the *Federal Register* prompted more than 20 replies, principally from persons in the American nuclear industry and state or city health departments. These letters are filed at the AEC Public Document Room, Washington, D. C., and are summarized in the June 15 and 22 issues of the *Atomic Energy Clearing House*.[10] With few exceptions the letters agreed to the need for site criteria, but varied considerably regarding the adequacy of the proposed criteria. The proposed criteria were criticized generally for being too vague. The most objection was directed toward establishing an exclusion radius and locating "large" reactors 10 to 20 miles from "large" cities. The proposed criteria of remoteness from populated centers cannot anticipate future population shifts, nor would they appear realistic in so far

as mobile reactors are concerned. Several letters noted that the real objective of these proposed criteria is to limit the off-site exposure; thus, if the exposure criteria were set, the reactor designer would be able to design for the optimum combination of exclusion radius and containment leakage for his particular reactor.

The letters from state health authorities frequently commented on the apparent omission of criteria for agricultural land, crops, and livestock, and also expressed interest in participating in site approval decisions.

It was stated in one of the letters that the chief purpose of the regulation was to establish the formula for financial protection. Whether or not this is true, it is apparent that the criteria for site selection and financial liability are closely related. The establishment of an exposure standard which the reactor plant must meet might then permit the assessment of financial liability on the basis of the surrounding population density and land value.

(W. B. Cottrell)

Safety in Experimental Systems

In the design and operation of reactors for testing materials and experimental systems, it is necessary to consider the effects of the experimental system on the reactor, as well as the hazards of the test reactor and the experimental system individually. A guide[11] to assist experiment sponsors in making hazards analyses of experimental systems for operation at the Materials Testing Reactor (MTR) and Engineering Test Reactor (ETR) has been prepared by the Phillips Petroleum Co. In this guide is given a brief comparison of the hazards pertinent to both reactor design and experiment design. Methods of calculation and means of obtaining design approval are not included; rather, the report serves to highlight the factors which must be considered in the design to obtain approval of experiment safety, including heat transfer, chemical reactions, and reactivity. Specific accidents are postulated and general protective design features, such as sample containment, are discussed. The report is brief and serves only as an introduction, with emphasis on safety, to the series of reports covering specific features of reactor experiment design of which it is a part. Other reports in the series are referenced.

An example of a hazards analysis of an in-pile experiment is the analysis of the KAPL-F-10 experiment[12] in the ETR. Analyses are made of the credible accidents, including the maximum credible accident (mca). The test system described (Report KAPL-35) is a pressurized-water loop. Its essential feature is a specially designed tube to fit into the F-10 facility of the ETR in which specimens of material, poison, or fuel may be tested at high neutron fluxes. The systems are designed as follows: a primary circulating system with a pressurizer; two auxiliary cooling systems (one forced flow and one convection flow) for cooling the in-pile tube; a water purification and sampling system; a primary water make-up system; a demineralized water system for removing heat from the primary system and for other cooling for which pure water is required; a utility cooling water system connected to the reactor secondary cooling water stream; a tube annulus system for maintaining dry air in the in-pile tube annulus which thermally insulates the inner tube from the ETR water; and the instruments, valves, pumps, and other auxiliary equipment required for the specified operations.

With the exception of the in-pile tube, the system was designed to the ASME boiler and piping codes. A mock-up of the in-pile loop was proof-tested for endurance under conditions more severe than predicted for operation.

Instrumentation alarms and controls are described in some detail from the standpoint of protection against malfunctions or inadequacies of heat transfer and hydraulic equipment, although no calculations are presented. Features for responding to other contingencies, including excessive radioactivity, are described in less detail.

The shielding was designed for maintaining the radiation at the outer surface below the maximum permissible levels during normal operation of the test loop. An accidental release of 1 per cent of the largest fuel specimen would result in a gamma-ray field of less than 10 r/hr at the outer surfaces of the cubicle shielding.

The maximum positive effect of nuclear coupling between experiment and reactor on $\delta k/k$ would occur during insertion of fuel specimens and is predicted to be less than 1 per cent. Various hazards are listed, including pump failure, pump power failure (one pump to be connected to a failure-free circuit), flow-channel blockage, temperature rise resulting from increased heat generation or loss of cooling capacity, pressure rise resulting from malfunction of the pressurizer, pressure loss as from crack in a container, and rupture of test specimen cladding. However, meltdown or rupture of the in-pile pressure tube is not stated to be a hazard. If failure of this tube were credible, pressure protection for the annulus would be required.

The maximum credible accident is considered to be a test specimen fracture that would release 5 per cent of the activity from a 1.5-Mw test specimen to the circulating loop. It is predicted that an operator would receive no more than a quarterly integrated dose if he left the area adjacent to the shielding wall within 10 min. A method of decontamination and/or removal of affected parts is described in which decontamination solutions, shielded casks, etc., are used.

(I. T. Dudley, D. B. Trauger, and L. A. Mann)

Fissionable Material Outside Reactors

EDITORS' NOTE: This section is comprised entirely of a paper prepared by the ANS Standards Project Subcommittee on Fissionable Material Outside Reactors and was presented by H. C. Paxton, Los Alamos Scientific Laboratory, at the 1959 Annual Meeting of the American Nuclear Society. It is included in its entirety here, inasmuch as it will not appear in the transactions of that meeting.

Standards for criticality control must be based on firm criticality data, and such data are incomplete though accumulating. As a consequence, standards in this field must be a matter of successive approximations, at first incomplete, or a strange mixture of realism and extreme conservatism, afterward to be filled in as more information becomes available.

The Nuclear Safety Guide,[13] which presents the combined efforts of a number of persons and organizations interested in this problem, is being considered for adoption as a standard by the American Standards Association (ASA), even though deficiencies are recognized, and revision is planned by an ASA subcommittee under the leadership of Frank Woltz of Goodyear Atomic Corporation. A major difficulty is the selection of concise rules that are sufficiently general to be useful without having unnoted pitfalls.

The manner in which tabulated limits of the concentration of fissionable materials listed in the Nuclear Safety Guide are obtained from experimental data is illustrated in Fig. 1. In this case the mass limits blocked out are for isolated, homogeneous, hydrogen-moderated U^{235} spheres with and without a water reflector, and they apply to cases where double-batching can be ruled out (note the limitations). The corresponding curves for Pu^{239} and U^{233} have much less experimental support. Other types of data, such as shape factors and density or dilution exponents, lead directly to limit increases where applicable.

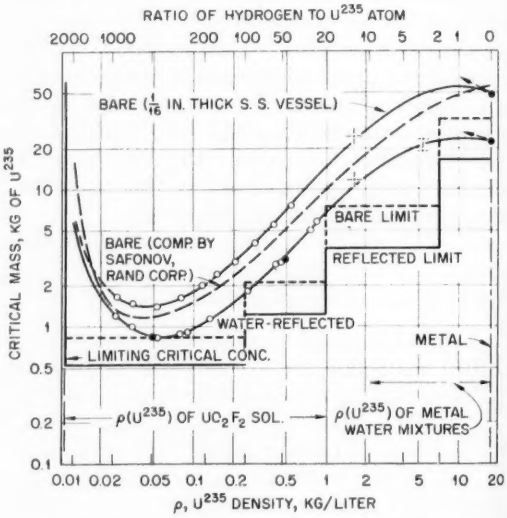

Figure 1—Critical masses of homogeneous water-moderated highly enriched uranium (93.2 per cent U^{235}) spheres. ●, direct spherical data. ○, from cylindrical data. ⊣⊢, from "hydride" data. Terminal slopes are from reactivity coefficient data.

Limits for simple isolated systems are only a starting point, for isolation and simplicity

are seldom found in a processing plant. The Guide, therefore, deals with interaction and reduction of complex systems to conservatively equivalent cylinders or spheres. It also gives rules for storage and transportation.

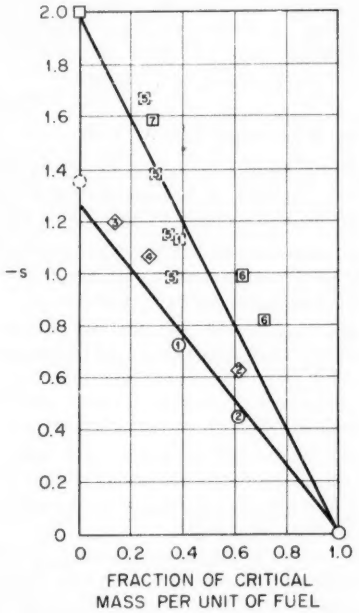

Figure 2—Storage capacity of three-dimensional lattice. ○, concrete-reflected lattice. ◊, partially reflected lattice. □, unreflected lattice. (Data from J. T. Thomas and C. L. Schuske, LASL, are included.)

1: Equivalent 20-kg, solid, highly enriched uranium spheres

2: Equivalent 32-kg, solid, highly enriched uranium spheres

3: 24.5-kg, solid, highly enriched uranium slabs, 1 × 8 × 10 in.

4: 30-kg assemblies of ~5-kg highly enriched uranium pieces with an effective density of 40 per cent

5: Highly enriched UO_2F_2 solution in 6-in.-diameter cylinders; hydrogen-to-U^{235} ratio of 44

6: Highly enriched UO_2F_2 solution in 8-in.-diameter cylinders; hydrogen-to-U^{235} ratio of 44

7: Highly enriched UO_2F_2 in three 3-in.-thick, 47.5-in.-wide slabs of variable height; hydrogen-to-U^{235} ratio of 337

s = Lattice-density exponent of the expression F^{-s}, where F is the volume fraction of the lattice that is occupied by the indicated unit

The specific rules for storage and transportation seem to be over-restrictive for materials of large critical volume, such as slightly enriched uranium metal or dilute solutions. The rules were based on experimental arrays of small-volume metal units, and generalization to much larger units was believed at the time to be risky. Data now exist, some very crude, on a sufficient variety of lattices to test generalizing schemes.

For three-dimensional arrays of a given unit, it appears that the storage capacity of the lattice (say, in number of spherical critical masses of material) is proportional to F^{-s}, where F is the volume-fraction of the lattice that is occupied by units and s is a constant that may be called the lattice-density exponent. The way in which s correlates with the reactivity index of a unit (in this case fraction of critical mass) and with degree of reflection about the lattice is illustrated in Fig. 2 for a considerable variety of unit types. If this sort of correlation is considered convincing, storage and transportation limits may be generalized therefrom immediately.

The types of data that exist for linear and planar storage arrays are shown in Fig. 3. This is only one example of efforts to fill gaps in the potential standards for criticality control.

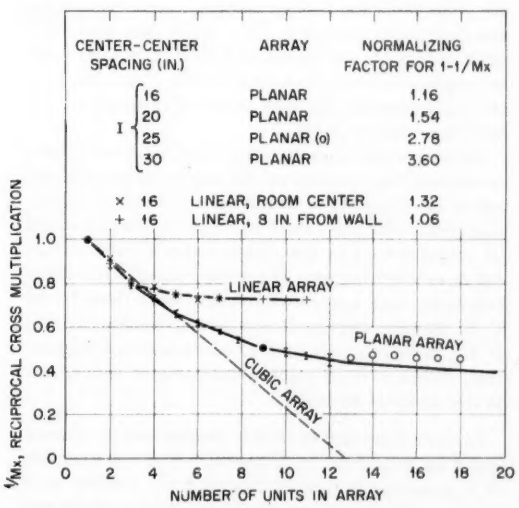

Figure 3—Linear and planar arrays of approximately 32-kg highly enriched uranium spheres centered 8 in. above concrete floor in room with concrete walls.

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*Appendix: Notice of Proposed Rule Making**

The Commission is considering the formulation of an amendment to its regulations to state site criteria for evaluation of proposed sites for nuclear power and test reactors and is publishing for comment safety factors which might be a basis for the development of site criteria.

In view of the complex nature of the environment, the wide variation in environmental conditions from one location to another and the variations in reactor characteristics and associated protection which can be engineered into a reactor facility, definite criteria for general application to the siting problems have not been set forth.

All interested persons are invited to submit comments and suggestions on the following site factors and on development of definitive criteria for evaluation of sites for power and test reactors which might be incorporated in the Commission's regulations. All interested persons who desire to submit written comments and suggestions should send them to the U. S. Atomic Energy Commission, Washington 25, D. C., Attention: Division of Licensing and Regulation, within 30 days after publication of this notice in the *Federal Register*.

Factors Considered in Site Evaluation for Power and Test Reactors. a. *General.* The construction of a proposed power or test reactor facility at a proposed site will be approved if analysis of the site in relation to the hazards associated with the facility gives reasonable assurance that the potential radioactive effluents therefrom, as a result of normal operation or the occurrence of any credible accident, will not create undue hazard to the health and safety of the public.

There are wide possible variations in reactor characteristics and protective aspects of such facilities which affect the characteristics that otherwise might be required of the site. However, the following factors are used by the Commission as guides in the evaluation of sites for power and test reactors. The fact that a particular site may be deemed acceptable for a proposed reactor facility when evaluated in the early phases of the project, does not determine that the reactor will eventually be given operating approval, or indicate what limitations on operation may be imposed. Operating approvals depend on detailed review of design, construction and operating procedures at the final construction stages.

b. *Exclusion distance around power and test reactors.* Each power and test reactor should be surrounded by an exclusion area under the complete control of the licensee. The size of this exclusion area will depend upon many factors including among other things reactor power level, design features and containment, and site characteristics. The power level of the reactor alone does not determine the size of the exclusion area. For any power or test reactor, a minimum radius on the order of one-quarter mile will usually be found necessary. For large power reactors a minimum exclusion radius on the order of one-half to three-quarter miles may be required. Test reactors may require a larger exclusion area than power reactors of the same power.

*Reprinted from Federal Register, May 1959.

c. *Population density in surrounding areas.* Power and test reactors should be so located that the population density in surrounding areas, outside the exclusion zone, is small. It is usually desirable that the reactor should be several miles distant from the nearest town or city and for large reactors a distance of 10 to 20 miles from large cities. Where there is a prevailing wind direction it is usually desirable to avoid locating a power or test reactor within several miles upwind from centers of population. Nearness of the reactor to air fields, arterial highways and factories is discouraged.

d. *Meteorological considerations.* The site meteorology is important in evaluating the degree of vulnerability of surrounding areas to the release of air-borne radioactivity to the environment. Capabilities of the atmosphere for diffusion and dispersion of airborne release are considered in assessing the vulnerability to risk of the area surrounding the site. Thus a high probability of good diffusion considerations and a wind direction pattern away from vulnerable areas during periods of slow diffusion would enhance the suitability of the site. If the site is in a region noted for hurricanes or tornadoes, the design of the facility must include safeguards which would prevent significant radioactivity releases should these events occur.

e. *Seismological considerations.* The earthquake history of the area in which the reactor is to be located is important. The magnitude and frequency of seismic disturbances to be expected determine the specifications which must be met in design and construction of the facility and its protective components. A site should not be located on a fault.

f. *Hydrology and geology.* The hydrology and geology of a site should be favorable for the management of the liquid and solid effluents (including possible leaks from the process equipment). Deposits of relatively impermeable soils over ground water courses are desirable because they offer varying degrees of protection to the ground waters depending on the depth of the soils, their permeability, and their capacities for removing and retaining the noxious components of the effluents. The hydrology of the ground waters is important in assessing the effect that travel time may have on the contaminants which might accidentally reach them to the point of their nearest usage. Site drainage and surface water hydrology is important in determining the vulnerability of surface water courses to radioactive contamination. The characteristics and usage of the water courses indicate the degree of risk involved and determine safety precautions that must be observed at the facility in effluent control and management. The hydrology of the surface water course and its physical, chemical and biological characteristics are important factors in evaluating the degree of risk involved.

g. *Interrelation of factors.* All of the factors described in paragraphs b through f of this section are interrelated and dictate in varying degrees the engineered protective devices for the particular nuclear facility under consideration, and the dependence which can be placed on such devices. It is necessary to analyze each of the environmental factors to ascertain the character of protection it might afford for operation of the proposed facility or the kind of restrictions it might impose on the proposed design and operation.

Simulator Studies of Reactor Accidents

The analog computer, an effective tool for examining the dynamic behavior of complex and nonlinear systems, is especially useful in studying the effects of malfunctions of reactor control systems. The information obtained is most worth while if the studies are performed before the accident occurs in actual operation. Such analyses have been made to determine the severity of operational accidents postulated for two pressurized-water reactors.

The effects of introducing cold water into the Shippingport Pressurized Water Reactor (PWR) by two different mechanisms were analyzed by d-c analog simulation to determine the minimum conditions beyond which safety limits were exceeded.[1] The first case involved the shutdown of the coolant pump in one of the four coolant loops which was then, through simulated "error," restarted at full speed after a shutdown time of 2 min. The water would be approximately 40°F below the average temperature of the coolant system and would thus subject sections of the reactor to a temperature change at a rate of 500°F/sec (based on known performance).

The second case involved operation of three coolant loops at full reactor power, with the fourth loop having been refilled with 70°F water after maintenance. Through operator "error" and because of hydraulic valve interlock "failure," the cold loop was connected into the reactor coolant system and the pump was run to full speed. This sequence of events would also cause sections of the reactor to be subjected to a much faster rate of temperature change.

The series of simulator calculations took into account inlet plenum mixing, hot-channel temperature deviation from the average temperature, uncertainty in the sectional temperature coefficient of reactivity, transport effects on heat transfer, and the limit of simulation validity at inception of bulk boiling. The use of two values of the sectional values of the temperature coefficient of reactivity and of the consequences of momentary overshoots of the blanket and core fuel temperatures produced variations in the coolant inlet temperature of 14 to 40°F, respectively. On the basis of the simulation results and metallurgical considerations of possible damage, a maximum allowable variation in the coolant inlet temperature of 20°F was chosen. The temperature factors that indicated this maximum were the blanket-fuel-rod UO_2-pellet melting point and the transformation temperature at which swelling of the uranium-zirconium alloy fuel plates would occur.

The report[1] on these simulator studies presents a comprehensive simulation and data analysis of a well-defined safety problem, but no control system design or operating procedure recommendations are made. (F. P. Green)

A general purpose d-c analog computer was also used to determine the consequences of the loss of coolant flow in the pressurized-water reactor being built by the Yankee Atomic Electric Company, Rowe, Mass. However, unless the reader is skilled in computer use, he will find that the mechanics of carrying out the analyses are difficult to understand. In the report[2] on this study the derivations of the equations used, as well as the nomenclature and some of the values of the parameters, are given in the appendixes. Since it is rarely possible in a study of a system as complex as this power plant to deal with all the physical phenomena which have a bearing on the problem under investigation, statement of the assumptions made is imperative. Some of the assumptions were necessitated by the limitations of the computer equipment. Others were made because of lack of experimental data in some areas. Since the basic assumptions and the conclusions constitute the major worth of the report,[2] both are quoted here, in part.

> The physical mechanism of the accident and the related thermal and neutron kinetics of the reactor core are as follows. As coolant flow decreases, the core outlet temperature increases and the film heat transfer coefficient decreases. This increase in average coolant temperatures causes a decrease in reactivity as a result of the negative temperature coefficient of reactivity associated with the

moderator-coolant region. A decrease in reactivity causes a reduction of reactor power level. The specific power of the reactor, the fuel pellet diameter, and the fuel thermal conductivity are such that a high thermal gradient is established in the fuel when the reactor is at full power. Hence, the average fuel temperature is well above the average coolant temperature. When a loss of coolant flow occurs, this large amount of heat energy stored in the fuel tends to maintain a relatively high heat flux at the moderator-clad interface while the film heat transfer coefficient is decreasing. These two effects result in a rapid approach to a burnout condition within some region of the core . . . .

This study deals with the transient behavior of the neutron and thermal kinetics of the core for the time interval after complete loss of coolant flow has occurred. System components external to the core such as the pressurizer and steam generator are not included in the study. All physical properties of the core materials were assumed to be constant and, consequently, the data presented in this report have full significance only before the time at which bulk boiling occurs. This temperature dependence has been omitted because of the lack of knowledge of the flow behavior during bulk boiling. A further limitation on the results arises from the fact that consideration of both flow redistribution and the effect of voids on the neutron kinetics was omitted . . . .

The assumption was [also] made that no thermal damage to the reactor core would be experienced if no bulk boiling occurred at the outlet of the core hot channel at any time during the accident. The elapsed time at which [boiling] occurred is also of importance since the heat flux at the surface of the clad has decreased with time. The longer the time required to reach bulk boiling, the lower the maximum heat flux and the less the likelihood that burnout would occur. The time to reach bulk boiling is thus an additional criterion for evaluation of results . . . .

The results obtained from this study show that for the 392-Mw(t) Yankee core with complete loss of coolant flow, bulk boiling occurred at the outlet of the hot channel approximately one second after the initiation of the accident for the condition of the instantaneous scram. As a result of the relatively low thermal conductivity and the large heat storage of the fuel material, the effect upon the outlet temperature caused by scramming the reactor is negligible for this initial portion of the transient. When stored mechanical energy is added to the primary loop, the flow decay rate is slower with the result that the increase in hot channel temperature is also much slower. When the reactor is scrammed, the increase in hot channel temperature can be kept below the bulk boiling point provided, of course, that sufficient energy has been added to the system.

Addition of mechanical energy to the loop might be in the form of increased pump inertia. Practical considerations for storing such mechanical energy undoubtedly would provide the limitation to this attempted solution of the problem of bulk boiling on simultaneous loss of all pumps.

The assumption that all four pumps could stop simultaneously in this power plant easily provokes arguments over how safe to make a reactor. Justification of the assumption was not an object of the analysis, but a decision must have been made regarding maximum safety since no mention was made of a situation in which the safety system was assumed to be inoperative during the pump failure transient. Failure of the safety system when needed is probably an unlikely event, and, yet, establishment of three independent power supplies for the four primary coolant system pumps is a feature of the safety system. Evaluation of the safety of a nuclear-power plant involves evaluation of the safety provided by each feature. It is therefore difficult to understand why this one feature of the safety system can be thought of as failing when, at the same time, little concern is shown over the failure of the scram system. (E. R. Mann)

Analysis of Reactor Transients

Several recent studies of reactor kinetics include observations of the pressure and temperature of a small homogeneous system following a rapid reactivity increase;[3,4] a stability analysis of a large test reactor, the ETR, with reactivity feedback through the temperature coefficient;[5] a correlation of data from power-burst experiments on five different reactors;[6] and studies of the thermal-transient behavior in both the PWR and a hypothetical water reactor.[7,8] In many instances the effect of the temperature coefficient and/or void formation has been shown to limit the reactor excursion and to thus prevent an accident. General agreement has been found among the results of several transient experiments and the results of calculations based on simple analytical models.

As a part of the Kinetic Experiments on Water Boilers (KEWB) program, experimental results have been obtained for small, aqueous homogeneous reactors and reported by Stitt and others[3] of Atomics International. The reactor consisted of a 12¼-in.-diameter stainless-steel sphere

(0.22-in. wall) having a capacity of 13.6 liters (exclusive of neck), surrounded by a 54-in. cube of graphite. Reactivity was introduced suddenly by rapid withdrawal of a centrally located poison rod. A record was made of power level, pressures at the bottom of the core and in the gas space above the free surface of the liquid, and temperature in the liquid, with the fuel solution (11.5 liters of uranyl sulfate) initially at a temperature of 25°C and a pressure of 15 mm Hg. The injection of 4 per cent reactivity resulted in a stable period of 2 msec. About 4 Mw-sec of energy was released as the fuel solution expanded, with the peak power being 530 Mw; the pressure at the top of the core peaked at 370 psi at 3.9 msec after the peak power was reached. Experiments started at initial pressures of either 43 or 71 mm Hg resulted in peak powers about 50 per cent greater than those observed at 15 mm Hg; higher initial temperatures resulted in correspondingly lower peak powers.

In further studies of water-boiler reactor excursions, Hetrick[4] reports that the pressure rise is about three times larger when the core is initially full of liquid fuel; and, also, that when the reactivity is added linearly in time, the peak pressure correlates well with the minimum period observed and agrees with the pressure-period correlation of Stitt and others.[3] Using this information, Hetrick concluded that the peak pressure occurring in the Armour Research Reactor after sudden insertion of 2.5 per cent reactivity would not rise above 300 psi.

(L. G. Alexander)

The stability of the ETR was examined by Gossman,[5] primarily to investigate its stability at full power (175 Mw) under design conditions. Measurements were made of feedback-frequency response, open-loop response, and closed-loop response; the responses were analyzed for stability by the Nyquist criterion. In the analysis it was considered that cooling water passed through the reactor core and then through the primary side of a heat exchanger and reentered the core. The temperature of the primary cooling water leaving the heat exchanger was used to control the coolant flow through the secondary side of the heat exchanger. The secondary coolant-flow–control system characteristics were based on experimental measurements.[5] The core and the heat exchanger were essentially considered to be "point" quantities (zero transit times).

An average negative temperature coefficient of reactivity was considered for the core, with the average core temperature being the linear average of the core-inlet and core-outlet temperatures. Since the core-coolant transit time was short, consideration of point values for the core should be valid. Inside the core, heat transfer from the fuel element to the coolant was considered by taking into account energy accumulation within the fuel element and the energy transfer from the element to the fluid. Since the heat-transfer coefficient associated with energy transfer from fuel element to coolant was high, 9000 Btu/(hr)(sq ft)(°F), the relation between the fuel element temperature and the coolant temperature under equilibrium conditions may have been sufficient rather than the time-dependent relation that was used. In the heat exchanger, heat transfer was based on the linear average of inlet and outlet temperatures. Since the transit time in the heat exchanger would be significant, a better mathematical representation of the heat exchanger would be warranted if the heat-exchanger behavior were found to influence the results. The conventional linearized reactor kinetic equations were used to relate reactor power to reactivity; six delayed-neutron precursors were considered.

The ETR was found to have a high degree of stability on the basis of its negative temperature coefficient of reactivity. The temperature-activated controller for effecting changes in secondary coolant flow was, however, found to be ineffective. Because of the slow response associated with the heat exchanger, it is doubtful whether stability can be significantly affected by secondary coolant-flow control except at very low frequencies. (P. R. Kasten)

Power-burst experiment data from SPERT, KEWB, BEPO, BORAX, and GODIVA have been well correlated by Corben,[6] who shows that some of the behavior observed may be derived theoretically by an analysis that does not make any reference to the shut-off mechanism and, in some cases, is indifferent to the details of the reactor kinetic equations. Corben predicts that in excursions having a period τ less than 10 msec, the maximum power P_m is, to a good first approximation, equal to

$$\frac{1}{2} P_o \exp (t_m/\tau)$$

where P_o is the initial power, and t_m is the time

elapsing between insertion of reactivity and attainment of maximum power. Using this relation, he obtains three others, namely,

$$\tau P_m = \frac{2}{3} E(t_m)$$

where $E(t_m)$ is the integrated energy release at time t_m;

$$E_{tot} = 2P_m$$

where E_{tot} is the total energy released in the excursion; and

$$B = 1.76\tau$$

where B is the burst width (at half-maximum power) in milliseconds. These approximate relations correlate the available data with considerable success.

Wieczorkowski[7] has studied thermal-transient behavior in the PWR (both at power and in the shutdown conditions) by means of an analog computer. Variables included were coolant-flow rates, inlet temperature, initial power level, rod worth, and withdrawal rates. The results were presented in the form of plots of peak power, peak coolant temperature and pressure, peak hot channel temperature, and coolant volume surge. Typical "worst case" results are shown in Table II-1.

Burnout occurs when the burnout ratio falls below 1.5. It was concluded that the reactor was safe from burnout for the transients studied and that boiling in the core would not occur. The greatest danger was felt to arise from large volume surges that might occur when the reactor was started up with the coolant temperature and pressure below normal for operation. The resulting pressure surge might damage the system before the pilot-operated relief valve opened. (L. G. Alexander)

A widely used method for specifying the degree of safety of a reactor is to investigate the behavior of the system following a sudden introduction of 2 per cent excess prompt reactivity. The relevance of this behavior to the question of safety is somewhat doubtful, but, at least, a well-defined problem is presented. In a paper by Golian and others,[8] an attempt is made to give a rather complete discussion of the problem of temperature distribution during such an excursion within a reactor core which consists of alternate fuel, cladding, and coolant regions bounded by parallel planes. This is not an unduly idealized geometry since, as they point out, for the short period involved in the safety calculation, coolant motion is unimportant. Most existing and proposed fuel elements are assemblies of plates.

More specifically, a repeating structure is assumed that is characterized by a fuel, cladding, and coolant thickness. It is assumed that the heat from the source increases exponentially with time and is constant within the fuel regions. Starting transients are largely ignored, as is the departure from exponential behavior finally produced by shutdown, the latter consideration being completely in the spirit of the usual safety calculation. A reasonable fuel alloy is assumed in a single thickness. Three claddings are considered (aluminum, stainless steel, and zirconium), each in three thicknesses. The coolant is water and is assumed infinite, which should introduce little error.

Table II-1 WORST CASE TRANSIENTS IN PWR
(Data taken from reference 7)

Cause of transient	Rod withdrawal during power	Startup on four loops with pumps at full speed	Startup on three loops with pumps at half speed
Start-up rate		25 decades of power/min	25 decades of power/min
Reactivity inserted	$15 \times 10^{-4}\ \delta k$		
Initial power, %	100	1	1
Initial average coolant temperature, °F	523	250	250
Initial pressure, psia	2000	400	400
Peak power, %	125	160	74
Peak hot channel temperature, °F	606.5	375	425
Pressure at peak power, psia	2080	511	469
Saturation temperature at peak power, °F	639	469	460
Burnout ratio at peak power	2.28	2.25	2.3

Considerable detailed information was calculated on the important aspects of the temperature distribution for periods up to about 20 msec. Up to the time when boiling occurs, the calculations are straightforward, although laborious, and results are given for the peak fuel temperature when the coolant reaches boiling. On the assumption that shutdown is achieved by void formation, calculations are given for the peak power and the peak fuel temperature under various conditions. A considerably simplified assumption has been made for the heat conduction across the vapor film, but comparisons between calculated and measured peak power versus period and degree of subcooling for the BORAX experiments seem to justify the procedure used. (T. A. Welton)

Collapse of Parallel-plate Fuel Assemblies

Theoretical relations have been derived which are suitable for the approximate prediction of critical coolant-flow velocities for collapse of long, uniformly curved, parallel-plate fuel assemblies in the ETR.[9] At the critical velocity, modifications of the local linear velocity resulting from plate deflection produce a pressure differential sufficient to maintain the deflection. Important assumptions on which the analyses are based include uniform flow of an incompressible coolant, zero inter-channel leakage, and rigid side plates or supports. The results of the study are summarized as follows:

CASE A. Critical coolant flow velocity, V_c, for flat plates with fixed edges

$$V_c = \left[\frac{15\, gEa^3h}{\rho b^4 (1-\nu^2)}\right]^{1/2} \qquad (1)$$

If the fuel meat is assumed to have no shear or bending stiffness, Eq. 1 becomes

$$V_c = \left[\frac{30\, gEt_c^3 h}{\rho b^4 (1-\nu^2)}\right]^{1/2} \qquad (2)$$

CASE B. Flat plates with simply supported edges

$$V_c = \left[\frac{5\, gEa^3h}{2\,\rho b^4 (1-\nu^2)}\right]^{1/2} \qquad (3)$$

CASE C. Flat plates with simple supports spaced longitudinally (at intervals long in comparison with plate width)

$$V_c = \left[\frac{\pi^4\, gEa^3h}{48\,\rho l^4 (1-\nu^2)}\right]^{1/2} \qquad (4)$$

Equation 4 gives the lowest critical velocity corresponding to the simplest mode of deflection.

CASE D. Curved plates with fixed edges (for $a \ll R$)

$$V_c = \left[\frac{2\, gEa^3h\,\beta_1 \sin^5\alpha}{3\rho b^4 (1-\nu^2)\left(\frac{1}{6} - \frac{\sin 2\alpha}{8\alpha} + \frac{\cos 2\alpha}{12}\right)}\right]^{1/2} \qquad (5)$$

where $\beta_1 = (AR^2/I)\, f_2(\alpha) + f_1(\alpha)$

$$f_1(\alpha) = 2\alpha + \sin 2\alpha/8 \sin^2\alpha$$

$$f_2(\alpha) = f_1(\alpha) - 1/2\alpha$$

CASE E. Curved plates with hinged edges

$$V_c = \left[\frac{4\, gEa^3h\beta_2 \sin^5\alpha}{3\,\rho b^4 (1-\nu^2) C}\right]^{1/2} \qquad (6)$$

where $\beta_2 = (AR^2/I)\, f_3(\alpha) + f_1(\alpha)$

$$f_3(\alpha) = \left[\frac{4\alpha - 3\sin 2\alpha + 2\alpha\cos 2\alpha}{8\sin^2\alpha}\right]$$

$$f_1(\alpha) = 2\alpha + \sin 2\alpha/8 \sin^2\alpha$$

$$C = \frac{2}{3} + \sin 2\alpha\left(\frac{3}{4}\cot\alpha - \alpha\right) - \frac{4}{3}\cos 2\alpha + \alpha\cot\alpha\left(1 - \cos\frac{2\alpha}{2}\right)$$

In these expressions

a = plate thickness
b = width of flat plate or chord of curved plate
g = gravitational acceleration
h = initial flow-channel thickness at midspan
I = moment of inertia of beam cross section per unit beam width
l = longitudinal distance between plate-edge supports
E = Young's modulus of elasticity

α = one-half of angle subtended by a curved-plate arc between supports
ν = Poisson's ratio
ρ = weight density of coolant
A = area of beam cross section per unit beam width
R = initial mean radius of curvature of a curved plate
t_c = clad thickness on each surface

It is pointed out that membrane compressive stresses will decrease V_c to below the calculated value. Such a transverse loading could arise if the pressure on the exterior of the side plates exceeded the pressure in the channels between fuel plates. Conversely, transverse tensile loads on all plates would increase V_c to above the calculated value. Further uncertainties which could cause the actual V_c to differ from the calculated V_c include deviation of channel geometry parameters from nominal values, thermal stresses in regions of large temperature gradients, and possible modes of creep collapse arising from alteration of fuel-plate mechanical properties due to irradiation.

Flow tests should, accordingly, be conducted at velocities significantly greater than the design value for determination of the safety margin available. In particular, some allowance must be made for deleterious in-pile conditions, such as fuel burn-up.

In some fuel-plate assemblies, collapse has occurred at one-half the calculated critical velocity. For the ETR, for example, the calculated critical velocity with cold water is 54 ft/sec; but the actual average velocity at which fuel plates buckled in room-temperature tests, as reported by Beck and others,[10] was as low as 29 ft/sec. This result is attributable in part to deviations from initial nominal geometry and in part to a 20 to 25 per cent velocity difference between instrumented channels. Further discrepancies between ETR design values and experimental results have been noted[10] in experiments in which 10 of 930 fuel-element coolant channels and 16 of 256 control-rod coolant channels were instrumented. The core pressure drop required to establish an average coolant velocity of 35 ft/sec in the fuel-element coolant channels was 55 psi, which is to be compared with a design value of 42 psi; the pressure drop required for the same velocity in the control-rod coolant channels was 64 psi, which is to be compared with a design value of 42 psi.

The ETR fuel-plate collapse velocity was increased substantially by vertically slotting the side plates to equalize lateral pressure differences; these modified assemblies did not buckle at an average velocity of 39 ft/sec. A report containing detailed results of these tests has been issued.[11] (W. R. Gambill)

Leak in the Homogeneous Reactor Test Core Tank

On Apr. 4, 1958, following runs in which a total of 390 Mw-hr of thermal energy was generated, the fuel solution of the Homogeneous Reactor Test (HRT) was detected in the blanket region. A long narrow horizontal hole with a flow area of approximately 1.5 sq in. was found near the bottom of the core close to the junction of the 30 and 90 deg conical entrance sections,[12] and, in addition, a sample holder in the core was similarly damaged. The holes in the HRT core tank and sample holder appear to have been caused by metal temperatures above the melting point; the possibility that the holes resulted from corrosion or chemical reaction is inconsistent with the available information. Since simple heat-transfer considerations alone indicate that the melting point could not be attained without uranium deposition, estimates were made of the magnitude of uranium deposition required to melt a hole in the core tank.

In the subsequent analysis of the formation of the leak,[13] the physical model assumed that a uranium-bearing scale was deposited on the core-side surface of the core tank. Fissioning of this uranium generated heat that was transferred by conduction through the scale into the core fluid and through the core tank wall into the blanket fluid. The conditions under which melting of the zirconium core tank could take place were obtained by mathematically solving the appropriate heat-conduction problem. In estimating the surface temperatures of the scale and wall (blanket side), it was assumed that subcooled nucleate boiling took place, a condition that permits a good estimate of the surface temperature. If the heat flux exceeds that removable by nucleate boiling, the surface temperature could rise and film boiling could occur, which could lead to burnout at lower values of uranium deposition than estimated. No attempt was made to analyze flow conditions in the vicinity of the

hole or to postulate mechanisms of uranium deposition.

The maximum temperature of the core tank would occur at the wall-scale interface. It was assumed that when the interface temperature reached the melting point the condition existed for formation of a hole in the wall. The thickness of uranium-bearing scale required to raise the wall-scale interface to the melting point was calculated as a function of U^{235} density and effective diameter of the deposit. The results (for a 6-Mw operation) indicate that uranium deposits of relatively small area and thickness could result in melting of the tank wall. For example, a 1-in.-diameter deposit containing 1 g of U^{235} per cubic centimeter of scale need be only 100 mils thick to cause a spot on the wall to reach the melting point. With higher uranium concentrations the required scale thicknesses were from 30 to 60 mils.

The analysis considered the situation at the core wall. The same reasoning, however, applies to the sample holder and screens, and similar results are to be expected. Because of the many assumptions, the results should be considered only order-of-magnitude estimates. The estimates are of value from a qualitative viewpoint; that is, the estimated thicknesses of uranium deposits required to cause melting are small enough to seem possible and the required area is surprisingly small. These results emphasize clearly the danger associated with uranium deposition in areas of high neutron flux.

(H. C. Claiborne)

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Reactor Fuses

Control of a nuclear reactor is normally effected by the control rods which are motivated by signals from both the reactor control system and the reactor shutdown system. Although the control and shutdown systems normally operate with independent instruments and electronic equipment, the control rods in the reactor are frequently common to both systems. Thus failure of the control rods to insert (as would be possible in some instances because of thermal distortion of the core) would result in the loss not only of control but of shutdown capacity. In situations where such failure is deemed at all likely, another completely independent shutdown mechanism is provided. Such emergency shutdown systems may be divided into two classes, depending upon the provisions and occasions for introducing poison into the reactor. Pressurized-water reactors are commonly provided with means of adding boric acid to the primary coolant at the operator's discretion, whereas in other systems the poison addition system is fused so that the poison is automatically inserted into the reactor by some phenomenon associated with a reactor excursion. In consideration of the possibility that the over-all safety of a reactor might be enhanced by this latter type of device, considerable work has been done in recent years on different types of fuses.

Fuse tests conducted in the BORAX reactor are described in a report by Stilwell and Waterfield.[1] The fuse consists of a chamber containing helium under pressure that is separated from a container of powdered boron by a fission-heated, fusible rupture disk. The geometry of the fuse is such that expansion of the helium through the rupture disk will carry the boron powder into an annular space surrounding a fuel element in the reactor. A summary of the results which show reactor behavior with and without the fuse is presented in Table III-1.

Data on heat transfer and heat generation in fuse disks composed of a mixture of UO_2 in a lead-bismuth eutectic called "Cerrotru" were calculated. A strain-rupture curve was prepared, but the temperature at which the data were taken is not shown in the report.

Table III-1 RESULTS OF FUSE TESTS IN THE BORAX REACTOR[1]

Condition	Reactor period, msec	Max. power, Mw	Time to reach max. power, msec	Total energy released up to max. power, mw-sec
Without fuse	23.9	240	310	12.61
With fuse*	23.9	170	295	7.75
Without fuse	14.6	470	280	15.52
With fuse*	14.6	320	257.5	8.06

*Data for behavior with the fuse were estimated from the rupture-disk release time, and laboratory test data which show that 15 msec is required to insert the poison following rupture.

Triggers for igniting a gunpowder propellant to insert boron powder into a reactor have also been studied at General Electric.[2] Three types of triggers were examined: fusible rupture disks, bimetallic elements, and explosives consisting of U^{235} embedded in diazodinitrophenol.

The fusible disks were similar to those described above. The bimetallic elements were composed of uranium with mild steel and uranium with zirconium. Both straight and helical units were tested. The straight elements were arranged to provide overcenter snap action. Bimetal action was considered to be completed when an electrical contact was made. The straight bimetallic element with snap action gave superior performance, and tests showed that it operated from 52 to 81 msec before peak power was reached by the SPERT-I reactor on an 18-msec period. The report[2] includes a thermoelastic analysis of bimetal action and a description of the instrumentation used.

The developmental and experimental work carried on from 1954 to 1956 on three types of propellants used for insertion of nuclear poison (boron powder) into a reactor has been described by Spera, Stilwell, and Weiss.[3] These propellants were compressed helium, explosive gas (hydrogen or propane with air or oxygen), and solid

propellants (smokeless powder). Test setup provided a thin annular chamber into which the boron powder was propelled which was similar to the chamber used in the fuse test in the BORAX reactor.[1] Velocities from 50 to 975 ft/sec were reported. Different configurations of propellant and boron were investigated, and various annular volumes were used in the tests.

Some tests in which explosive charges were used to provide directed propulsion to insert poison into a reactor are discussed in another General Electric report.[4] The authors, Donaven, Stilwell, and Stuart, report that shaped charges in which the nuclear poison forms the liner will develop velocities of 6 to 20 km/sec. They state that a "shotgun" type of propellant system will insert poison into a reactor in less than 5 msec. These tests were intended to demonstrate and test a method of safely inserting poison. No design was proposed, and hence no conclusions with respect to reactor safety and operation can be drawn. Similarly, a thorough study of transient heat flow in the Atomics International Mark-II Reactor safety device, including both experimental and analytical work, has been described by Springer and Miller,[5] but no direct conclusions affecting reactor safety can be drawn from their study.

A fuse that is sensitive to power level has been described by Fitch and Springer.[6] This fuse, which was triggered by the fission-induced melting of a soldered plug or diaphragm which retains helium at 600 psi, was tested in SPERT-I. The release of helium pressure in turn released boron trifluoride at 1200 psi, that is, the device depended on the series operation of a fusible diaphragm and a rupture disk; it was called a "double-diaphragm" safety device. The device was self-contained and occupied the same volume as a standard SPERT-I element. The reactivity loss produced in the central region of the core was considerable. It is not clear whether the device that was tested represented the geometry required in a final design or whether it was dedevelopmental and peculiar to these particular tests. The tests amply demonstrated that under SPERT-I conditions the device was effective in removing the reactivity in excess of prompt critical as required to reverse transients in the 10-msec-period region. Use of the device also reduced the energy release and the peak power by factors of 10 and 25, respectively, in comparison with self-shutdown without the fuse. The tests also gave data in good agreement with calculations of thermal characteristics and with laboratory tests of poison release time, in which approximately 0.003 sec was observed as the time required, after fuse trip, for 50 per cent release of the boron. One fuse failed to function during a test because of improper assembly.

A major portion of the effort has been expended on the development of specific items necessary for a complete fuse assembly, and therefore no over-all conclusions as to the effect of fuses on nuclear safety can be drawn. Where tests of complete fuses are reported, they appear to be of a developmental nature, and many of the questions that would arise in assessing the merit of a particular fuse design in terms of over-all reactor safety and operation are not discussed. More information is needed in the following areas:

1. Statistical variations and tolerances to be expected in such devices.
2. Stress-rupture as affected by time and temperature where materials exhibiting these characteristics are used in critical fuse components.
3. Methods of nondestructive testing which will ensure that the device will operate as designed.
4. The general safety problem that is created when any device containing a void or potential void is inserted into a reactor.

The latter problem is aggravated when the void is a pressure vessel and when the untriggered device represents a large contribution to negative reactivity in the core.

The behavior of the devices at reactor periods of less than 10 msec must also be investigated. Most fuse operation is predicated on the integration of energy during a transient. As transient periods decrease, integrated energy decreases for a given rise in power. A region of particular interest is that of fuse operation in situations where the periods are shorter than 10 msec.

(J. R. Tallackson)

Reactor Safety System Design Criteria

EDITORS' NOTE: This section is comprised entirely of a paper prepared by Frank Larin, A. B. VanRennes, and Richard Olsen of the ANS Standards Project Subcommittee on Reactor Dynamic System Design and was presented by Mr. Larin at the 1959 Annual Meeting of the American Nuclear Society. It is included in its entirety

here, inasmuch as it will not appear in the transactions of that meeting.

The term "reactor safety system," as used here, means the neutron detectors, instrumentation, and reactor shutdown mechanisms required to measure reactor neutron-flux levels (and, generally, the time rate of change of the neutron flux), to determine whether the flux indications have exceeded selected trip levels, and to cause appropriate corrective action when the trip levels have been exceeded.*

The purpose of this report is to present a check list of design practices recommended for incorporation in reactor safety systems to give assurance of reliable operation even with failure of some portions of the system. The list is intended primarily as an aid to reactor designers and users when designing or purchasing equipment and for evaluation groups who have heard the term "fail safe" so often for so many different types of equipment that it no longer has a concrete meaning. The concern here is with design concepts. Many important performance requirements such as accuracy, response time, stability, reactivity insertion rates, and allowable excess reactivities will not be considered here since they fall outside the scope of this report.

In discussing failures in a reactor safety system, a temptation exists to simplify the problem by stating merely that a design should provide for reactor shutdown whenever a component fails, that is, by "failure-to-safety" or fail-safe design. In practice, it is not always possible to obtain the same output response for all categories of circuit or component malfunction (open circuits, short circuits, or grounds), or for changes in circuit parameters such as gain, resistance, capacitance, etc.

Furthermore, absolute failure-to-safety may not always be desirable, for two reasons. First, failure-to-safety for all component failures could cause serious inconvenience from inadvertent shutdowns. It may be preferable to sound an alarm and indicate the instrument malfunction. The second reason is psychological and closely related to the first. If the design is inferior but incorporates fail-safe features which cause a large number of unnecessary shutdowns, operating personnel may alter the equipment to keep the reactor in operation and compromise safe operation. Any equipment that can cause operator frustration is potentially dangerous. Fortunately, there are design methods which can alloy equipment failures without weakening the reactor integrity or shutting the reactor down.

Four primary methods are available in various combinations to assure reactor shutdown when necessary, and which can, at the same time, reduce the probability of inadvertent reactor shutdown. These design features are:

1. Fail-safe designs.
2. Use of monitoring and test circuits.
3. Use of redundancy in the form of identical parallel signal channels and coincidence circuits for comparison between these channels.
4. Use of highly reliable components.

Failure-to-safety should be exploited in prudent design of control circuits so that the most probable failures cause the output to indicate an abnormal condition. Often such design is accomplished by operating critical points off ground so that loss of power, shorts to ground, or open circuits will cause an abnormal signal. In other cases, current and voltage biases necessary to give a normal signal are obtained from the power supply through series connections which check circuit continuity. In any case, the design should attempt first to eliminate as many causes of failure as possible and, second, to give the proper abnormal response to the most important component failures.

Monitoring may be used to determine the proper operation of portions of control circuits. Such practices include continuous checking of voltages, currents, connections, superimposed a-c signals, outputs of other parallel circuits, etc. These check circuits should be as simple and reliable as possible. Careful thought should be given to the proper action desired in response to an indication of failure. Some of these alternatives will be discussed later.

In addition to monitoring, test signals may be introduced into the equipment to check for proper operation or for calibration. These may be developed by elaborate automatic test equipment or manually by simple voltage or current sources. Again, reliability of the test equipment is important; poor test equipment can create more troubles than it locates.

Redundancy is the combining of independent channels which measure the same variable and

*Although Mr. Larin stated orally that his paper referred specifically to swimming-pool type reactors, his analysis is applicable to other reactors.

initiate a common response, and it should always be used whenever a failure of one circuit might create a dangerous situation. Ultimately these redundant signals are combined at a common point or junction of the channels to create the desired output response. For maximum reliability, this common point should be located as far toward the response end of the system as possible so that the number of components common to all signal paths is minimized.

Two systems for joining channels are in use; in one system an excessive signal, or trip signal, from any channel will cause a safety response, and, in the other, trip signals are required from two or more channels to effect a safety response. The latter system, usually called a coincidence system, is inherently a digital function. Therefore, in each channel, threshold or bistable circuits must precede the common point of coincidence comparator to decide whether or not each signal is excessive. Besides preventing unnecessary scrams, a coincidence circuit has the advantage of permitting test of individual channels or trip circuits during operation without reactor shutdown.

The former system, in which any channel can cause a response, may inject either analog or digital signals at the common point. In such a system, the combining of analog signals so that the larger governs has in the past been termed "auctioneering."

The auctioneering function for analog signals is logically somewhat similar to a digital "OR" gate, which generates an output response whenever a signal is delivered to any of its inputs.

Combinations of fail-safe techniques with redundant and coincident measuring channels can be highly effective in providing reliable reactor control.

The use of circuit components designed for highly reliable operation and the application of conservative engineering design practices greatly increases the reliability of reactor-control systems. Improved long-lived vacuum tubes are now available for a large number of applications. Magnetic amplifiers are used where moderate speed of response and sensitivity are required and where waveform is not important. Transistor and diode reliability has been increased to the extent that their integrity now approaches that of passive circuit components. The use of such reliable components is paying dividends in reducing the requirements of monitoring, testing, failure-to-safety, and redundancy, and is thus decreasing equipment complexity. Simplicity of equipment further increases reliability.

The techniques used in reliable circuit design have been examined thus far; now, and more specifically, some criteria that may be used to evaluate a particular design will be presented. The criteria pertain primarily to power-level safety channels since most reactors require this protection for safe operation. However, they are also applicable in general to other information channels (that is, period, power derivative, temperature, pressure, etc.) that are necessary for safe operation. Necessary here means that failure to shut the reactor down when the variable exceeds selected limits could cause serious damage and not merely minor damage or inconvenience.

Since a reactor may be protected by a number of instrument arrangements, an instrument malfunction need not require the same action in every design. A general minimum rule will help clarify the action necessary for various failures: At least two independent channels which are necessary for safe operation must be fully operative or the reactor should be shut down.

In the criteria listed below, the term "trip the channel" may have a different meaning depending on the arrangement of independent channels and the application of the general rule requiring two operative channels. If there are two power-level safety channels, it means scram. If more than two channels are arranged in coincidence, the defective channel only is put in a condition calling for a scram. If there are more than two channels in a redundant arrangement, then it means that the circuit shall be interlocked to cause a scram whenever additional channel trips leave fewer than two operative channels.

A list of suggested criteria for reactor control systems follows:

1. At least two independent channels which are necessary for safe operation must be operative at all times.
2. There should be no unnecessary units common to independent channels.
3. Loss of ion-chamber supply power should trip the channel.
4. Loss of instrumentation power should trip the channel.
5. Loss of continuity of the interconnecting cables should trip the channel.
6. Disconnection of any component of an independent channel should trip the channel.

7. Open circuits, short circuits, and grounds in the electronic equipment should, where possible, trip the malfunctioning channel.

8. Means should be provided to indicate proper operation of the channel, including monitoring of operating conditions, insertion of test signals, detection of a-c signals imposed on the ion-chamber output, detection of minimum ion-chamber currents, etc.

9. Safety rods should not scram against gravity.

10. Safety rods should be designed so that no malfunction can permit the control rod to drop through the reactor core.

11. Safety rods should be designed so that a rod cannot inadvertently withdraw a fuel element from the core.

12. Failures of components in the control-rod system should cause either no change in reactivity or should introduce a decrease of reactivity in the reactor.

13. Independent channels are usually joined at a common point (compared) to provide a scram signal from any channel to all scram mechanisms. This common point where the channels are joined should be functionally as near the scram mechanism as possible.

14. Open circuits, short circuits, and grounds after the common point of the independent channels should cause reactor shutdown unless the malfunction does not prevent subsequent reactor shutdown.

15. Disconnection of any section of equipment common to all channels should cause reactor shutdown.

16. Any switches or adjustments capable of rendering a channel inoperative should be interlocked to trip the channel when in the nonoperate positions.

17. Introduction of test signals should not open the normal signal path unless a monitoring system is used to assure continuity of the switched connection between transducer and amplifier input circuit.

18. Malfunctions of test equipment should not prevent operation of the channel in a safe manner.

19. The system should be designed in a manner to assure location of the neutron detectors in a position where the flux has a relatively constant ratio to the average core flux to assure that this ratio cannot be changed by some means and to assure that the sensor is actually in the proper position.

20. Means should be provided for determining that an indication of neutron flux is indeed caused by neutrons and not by other radiation or noise. This identification is generally difficult and may have to be accomplished by operational procedures.

A problem of oversimplification exists in setting down criteria for evaluating the reliability of reactor control systems. Such criteria serve a useful purpose in pointing out some of the considerations of good design. They do not and cannot cover either the majority of criteria to be applied to a particular reactor or the most important considerations of reactor safety. The greatest increase in reactor safety comes from improvements in reactor design which place less reliance for safety on the reactor control system, such as favorable temperature, void, or power coefficients.

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Evaluation of Containment System

There have been a number of approaches to the design of reactor containment. The basis for selection of one type of container or another is generally a balance between minimum cost and control of the activity that would be dispersed in the event of the maximum credible accident (mca). The types of containment provisions usually considered may be divided roughly into three categories: (1) leaktight pressure-containment shells encompassing the reactor coolant system in which the coolant system enclosure is the primary container and the shell is the secondary container; (2) confinement structures, such as an enclosed building, into which activity is released as it is generated and exhausted through filters to a stack; and (3) self-contained fuel elements, which would release only small amounts of activity during the mca, surrounded by the coolant system enclosure, which provides secondary containment.

The first category is typified by the pressurized-water and boiling-water reactors; the second, by the Oak Ridge Research Reactor, the Hanford and Oak Ridge Graphite Reactors, and the British Windscale Reactors; and the third, by the British gas-cooled reactors at Calder Hall, Berkley, Bradwell, Hinkley Point, and Hunterston. Discussions of the criteria used in the evaluation of these types of containment provisions have been given in a number of recent papers.[1-9]

Experience gained in the United States in designing secondary-containment vessels has been summarized in the June 1959 issue of *Power Reactor Technology*.[10] The table in reference 1 also includes silhouettes of the containment vessels for a number of reactors.

The EBR-II fast-reactor containment shell has been described by Monson and Sluyter.[2] Explosion considerations are a particular factor in this design because of the possible reaction between the sodium coolant and the air in the containment enclosure. In this case the containment shell is an 80-ft-diameter cylinder with a hemispherical top and a semiellipsoidal bottom. The total height is 139 ft and 48 ft of the shell is below grade. A $26\frac{3}{4}$-in.-thick internal blast shield is provided between the reactor and the biological shield. The vessel is designed for an internal pressure of 24 psig, and its specifications are not greatly different from those for the containers for the boiling-water reactor types. However, because of the explosion potential from the sodium-air reaction, particular attention has been paid to the mode of energy release from the mca in this reactor concept. The following factors were of interest in arriving at the final design:

1. The principal shock wave from the explosion would be absorbed by the large volume of sodium coolant which surrounds the reactor. Only 10 per cent of the shock wave would reach the outer boundary of the sodium tank.

2. The free surface of the sodium would greatly reduce the shock forces directed at the top of the vessel (reportedly by about two orders of magnitude) since most of the explosion would propagate through the liquid sodium to the vessel bottom and sides.

3. The duration of the shock wave imposed on the top of the vessel from argon pressure and liquid sodium would be less than 0.002 sec.

4. Submersion of the major components in the bulk sodium tank would prevent them from becoming missiles.

The analysis showed that the containment structure is capable of withstanding a detonation force equivalent to 300 lb of TNT.

A condition of interest to the designers of the EBR-II containment structure was the temperature rise in the shell due to the heat generated from the sodium-air reaction. The shell is constructed of A201 carbon steel, which retains its strength at temperatures up to 650°F. Calculations were made to determine the temperature rise in the shell due to the dissipation of heat from the reaction, and it was shown that, even on the basis of extremely pessimistic assumptions, the shell temperature would not rise above 400°F. This result

will be of particular interest to people working on reactor systems in which exothermic reactions might release large quantities of heat to the containment enclosure.

The discussion by Bergstrom and Chittenden[1] brought out the fact that Section VIII of the ASME Boiler and Pressure Vessel Code, Unfired Pressure Vessels, has been the basis for all major reactor containment vessels built in the United States. The writers argued that re-evaluation of some of the requirements of the Code, as applied to containment vessels, would be appropriate for the following reasons:

1. The Code provides for continual application of the design stress, whereas containment vessels are required to withstand the design stress only "once in a lifetime." Thus lowering the factor of safety from 4 to 3.5 or 3 could be logical and justifiable.

2. In larger reactors the thicknesses required by the present Code require field stress relieving, which is probably impractical for a large vessel and thus limits the design.

3. The Code does not take into account stresses at discontinuities in the surface, and therefore, to allow for stress concentrations, areas subjected to pure membrane stresses must have the same thickness as that established by the stress concentration points.

4. Formulas for computing stresses in vessel openings have been developed mainly to apply to relatively small vessel openings and cannot be assumed to be conservative when blindly applied to the large vessel openings required in containment vessels.

Arnold of the Chicago Bridge & Iron Co.[3] has raised a question about preheat requirements for welds in connection with the Dresden containment vessel. For class P-1 materials $1\frac{1}{4}$ to $1\frac{1}{2}$ in. thick the Code requires a 200°F preheat of plates prior to welding. Tests were performed on plates which had been cooled to 30°F after being assembled in a manner which would apply lateral and angular restraint to the weld area. The welds were performed with low-hydrogen electrodes, and no deterioration of the notch toughness of the weld area was found as a result of the failure to preheat. The test results indicate that the 200°F preheat could be eliminated for vessels welded by similar methods from similar materials. Elimination of the preheat requirement would result in substantial cost savings and reduced fabrication complications.

Evaluations of pressure-containment vessel designs have been presented by Smith and Randolph of the Bechtel Corporation[4] in connection with the selection of the Dresden containment vessel design. The important differences between containment vessels and ordinary pressure vessels are pointed out to be the following:

1. Containment vessels are not normally under pressure.

2. Personnel access through locks must be provided to avoid impairment of containment integrity.

3. Ventilation air supply and exhaust must be provided.

4. Safety valves defeat the purpose of the vessel.

5. Missile shielding and leaktightness require special attention.

6. Many penetrations of the vessel are necessary.

Smith and Randolph discuss a number of alternatives considered for the Dresden design. The following criteria were used in the evaluation:

1. Ease of design and construction of the containment membrane.

2. Ease of design and installation of access locks and penetrations.

3. Accessibility for inspecting and testing for strength and tightness.

4. Predictability of the performance of materials which make up the containment membrane.

5. Economy in materials costs.

6. Ease of design and construction of the transition between the containment membrane and the foundation.

The term "containment membrane" has been used to encompass all types of containment enclosures. For the Dresden design, in addition to the now almost conventional pressure vessel finally selected, concrete-lined enclosures with metallic membranes and anchored domes and tanks were all given consideration in the design evaluation. The containment vessel finally selected was a 190-ft sphere. Spherical vessels have proved to be the most economical con-

tainment for boiling-water and pressurized-water reactors. A sphere is not the only method of containment, however, as evidenced by the others considered for Dresden.

The Dresden design is described as utilizing partial containment; the turbine generator is outside the containment vessel. It is worth noting that in the boiling-water design, of which Dresden is typical, the reactor coolant (light water) flows directly to the turbine outside the containment vessel. Therefore, unless provisions for valving off the coolant inlet and outlet to the reactor are provided, the turbine forms part of the containment envelope. The PWR, the EGCR, and most other reactor types have an intermediate heat exchanger between the reactor coolant and the drive steam for the turbine, and the intermediate heat exchanger is located inside the containment vessel. However, it should not be concluded from this design difference that the Dresden containment is less satisfactory than that used in other designs since the mode of failure anticipated in a mca is all-important in determining containment design requirements. The kinetics of the mca determine the methods by which containment integrity must be maintained.

Of significance also is the need for an internal heat sink to absorb heat generated during the mca and by fission-product decay. In the case of Dresden, isolation of the turbine removes the heat sink represented by the turbine condenser, but a water-spray system is provided to remove decay heat in this instance.

The Dresden designers placed particular emphasis on the ability to meet the ASME Unfired Pressure Vessel Code requirements, although it was not clear that they considered this essential to the containment membrane design. The particular containment structures considered were:

1. A 180-ft sphere (later increased to 190 ft) with the turbine-generator external to the sphere.
2. A 220-ft-diameter buried cylindrical vault with a sheet-metal liner and with steel hold-down columns.
3. A 220-ft-diameter partially buried cylindrical tank with steel hold-down columns.
4. A tunneled vault.
5. A capsuled vault to contain the reactor, with the vault being a steel tank with hemispherical ends encased in concrete.
6. A 220-ft-diameter steel hemispherical dome anchored in the rock excavation, with a metal membrane at the base.
7. A 215-ft-diameter partially buried capsule encased in concrete to contain the reactor.

Structures 1, 5, and 7 do not provide for enclosing the turbine; the other structures provide full containment.

The analyses of the various designs show each to present one or more problems, as described below:

1. The concrete enclosed tanks and the dome type designs will not meet ASME Code requirements.
2. If the containment membrane must be anchored in the rock foundation the rock must be capable of withstanding shear loads, and, for the Dresden location, the performance of the rock was unpredictable. It is apparent that a careful examination of the rock structure is required for such designs.
3. Where the containment vessel is embedded below grade, the possibility of buoyant forces from ground water must be considered. Surrounding areas must be sealed against ground water or the foundation must be designed to withstand the buoyant forces.
4. When a membrane must be supported directly by the foundation, pressure grouting is required, and there is concern about the effectiveness of pressure-grouting procedures.
5. Inspection and testing of the membrane when encased in concrete or rock impose a serious restriction on the design, because pressure testing cannot be performed without the encasement, and the encasement makes inspection impossible. As pointed out by Bergstrom and Chittenden,[1] a recent ASME Pressure Vessel Code ruling (Case No. 1228) permits preinspection and leaktesting of welded joints prior to placement of the concrete provided the complete vessel is ultimately given a pneumatic test under Code conditions. This ruling, which was passed after the Dresden plant was under construction, might have altered the design conclusions reached by the Dresden designers.
6. The use of a membrane held down by columns complicates the equipment layout substantially because of column interferences, and

there is inherent difficulty in anchoring the columns to withstand the high tensile loads. This problem is peculiar to designs in which large areas of membrane must be reinforced in this manner. Smaller installations, such as the aqueous homogeneous reactor at Oak Ridge, are contained by this method.

7. Penetrations through buried or concrete encased containment systems are difficult to accommodate.

8. Access for construction and fuel-handling complicates the containment design and tends to discourage compact arrangements which rigidly fix the construction sequence and allow little flexibility for design modifications.

9. Temperature gradients between the containment membrane and its foundations must be considered in the design.

10. For a given pressure-containing configuration, when pressure-volume considerations determine the vessel size, the weight of material required in the pressure vessel is independent of the size of the vessel. Thus there is the alternative of designing a small vessel for high pressures or a large vessel for low pressures. Although not mentioned in the Smith and Randolph discussions,[4] loads imposed by the structure also have to be considered, and they sometimes govern the thickness of the containment vessel. Most large vessels have had material thicknesses in the range of $\frac{3}{4}$ in. to 1 in. in order to permit the vessel plates used in fabrication to have some dimensional stability.

Chave and Balestracci[5] have described the containment vessel for the Yankee Atomic Electric power plant, which is a pressurized-water installation. The containment vessel is a 125-ft sphere $\frac{7}{8}$ in. thick supported by 16 pipe columns extending to eight concrete piers. Concrete shielding inside the vessel individually compartmentalizes the four coolant loops. Although the containment structure has a detached concrete liner which makes an effective missile shield, the designers do not consider this necessary. As was the case for the Shippingport reactor, ductile steel is used throughout the coolant system; this material is not capable of failure in a manner which would generate missiles that could penetrate the containment structure. The design considerations were essentially the same as those for Dresden, and the conclusions reached were parallel.

Methods for establishing leaktightness requirements for pressure-containment structures have never been well defined. Bergstrom and Chittenden[1] tabulated the tightness specifications for a number of containment vessels. They range from 0.1 to 0.76 per cent of the contained volume per day under design pressure conditions. No pattern is apparent in the specifications. Bergstrom and Chittenden recommend 0.1 per cent of the contained volume per day as the design leak rate, but this appears to be based mainly on the fact that this is the best that anyone has ever attempted to do. The accepted practice is to base the design leak rate on the consequences of an activity release resulting from the mca.

None of the published material presently available gives an account of the design requirements for shutoff valves, ventilation, and emergency exits for containment structures. Filters and monitors on exhaust stacks, which are intimately involved in maintaining the containment integrity, are also unevaluated. Similarly, the operating experience with existing containment structures is unreported for the reactors that are currently in operation. These factors have much to do with the evaluation of reactor containment structures.

The continuous exhaust of activity from the mca through stack filters has not been examined for many types of reactor. The Oak Ridge Research Reactor is designed to incorporate this containment principle, which has been described by Cole and Gill[6] and by Binford and Burnett.[7] The building is a steel frame structure with insulated metal-panel siding and an insulated and built-up metal roof deck to provide a water-tight seal. All joints on the inside of the building are caulked and sealed. No exterior windows are provided and doors are sealed with rubber gaskets. Buildings of this type are not designed to contain pressure but to assure that gas outleakage other than through the stack will not occur during an activity release. The following sequence of events would occur in the event of an activity release:

1. The building ventilation system would shut down.

2. An automatic damper would open an emergency duct.

3. A gas scrubber consisting of a packed tower sprayed by a 5 per cent caustic solution would absorb the released activity, principally iodine, and give a decontamination factor of 99.95 per cent.

4. Absolute filters would remove particulate activity down to 0.3 μ in size.

5. The scrubbed and filtered vent gas would be exhausted by a 5000-cfm fan to the stack.

The success of this inexpensive type of containment is dependent on the effectiveness of the scrubber in obtaining a high decontamination factor and the ability to maintain the exhaust rate high enough to avoid building up a positive pressure in the containment building. The ORR scrubber is particularly effective for iodine, but for other sources of activity, such as the noble gases or strontium, adsoption media might be necessary, and the permissible throughput for adsorptive materials, such as activated carbon, can, in some cases, require a prohibitively high pressure drop across the adsorber. In such cases it might not be possible to meet the criterion of no positive pressure in the containment building during an emergency.

The principle under which the self-contained fuel element can meet the reactor safety containment criterion has been studied briefly at ORNL.[8] The conditions under which the ORNL GCR-2 was considered to be adequately contained without a secondary containment structure are quoted below:

"The hazards philosophy adopted for the GCR-2 was that two independent and improbable events, i.e., simultaneous leaks in a large number of fuel element capsules and the coolant gas circuit, either of which is immediately apparent to the operator, are required to produce a hazardous situation."

It is fundamental to this containment concept that there be a good system of fuel-element leak detection since the reactor operator must be able to shut down the system before leaks become significant. It is also essential that the containment system be adequate to allow for the maximum number of fuel-element failures resulting from the mca. An analytical approach to the evaluation of this condition has been described[9] as part of a report setting forth a hazards evaluation of a 30-Mw, prototype, partially enriched-uranium-fueled nuclear power plant. In the evaluation the mechanism of failure from the mca was traced, and it was shown that the amount of activity released when a rupture of the primary coolant system occurred was less than that which would subject the bystander to the maximum personal dose limit accepted by the AEC. This method of analysis is adequate only when the mechanism of activity release is clearly understood, and the uncertainties of such an analysis must be given careful scrutiny.

The foregoing discussion illustrates the variety of considerations to be factored into containment evaluation. It is apparent that no standard method of containment can be established for all types of reactors. The following seem to be the major premises on which each system is based:

1. Pressure containment is necessary when large amounts of activity are released concurrently with large volumes of coolant fluid under pressure.

2. Confinement structures are applicable when the activity released can be effectively scrubbed from the vent gas and the quantity of vent gas is small enough to be readily exhausted by conventional exhaust fans.

3. Fuel-element self-containment is feasible only when it can be shown that failure of the primary coolant containment system will not lead to failure of encapsulated fuel elements that would release substantial quantities of activity to the surrounding atmosphere. In any case, a thorough understanding of the mechanism and kinetics of the maximum credible accident is essential in evaluating the containment design. (M. Bender)

Structural Integrity of Containment Vessels

In order to minimize or eliminate the hazards associated with a nuclear reactor excursion, the reactor vessel or the reactor vessel and its containment shell must be designed to withstand both the internal pressure associated with the energy release and the missile impact. Therefore, the basic information associated with vessel behavior in each case must be determined before optimum containment design can be achieved.

The successful containment of a nuclear reactor excursion in which a large internal pressure is developed depends upon the ability of the pressure vessel to absorb energy through the mechanism of extensive plastic deformation. The strain-energy absorption characteristics of a given vessel bear little or no relationship to the ability of the vessel to meet design operating conditions. This is true because the allowable design stresses are necessarily limited to values which fall below the yield point of the material. Because of these facts, Wise, U. S. Naval Ordnance Laboratory, is investigating the energy absorption potential of pressure vessels.[11]

The premise underlying the investigation is that, for a given energy release and release rate, an optimum containment design exists. The design parameters are material, configuration, constraint, and size. The strain-energy equation of dynamic equilibrium for an internally loaded, hollow right-circular cylinder closed at the ends with rigid disks which provide rigid radial end restraints was obtained by considering the total energy in the system. The energy equations were simplified for the case of a homogeneous and isotropic cylinder subjected to a rotationally symmetric loading, and they were written in terms of experimentally obtainable transient loading and structural response parameters.

An experimental program was established in order to obtain the quantities required for calculating the strain absorption of a given shell. The quantities determined experimentally were the pressure bearing upon every internal differential area of the solid system and the radial and longitudinal coordinates of each such area. Both the pressure and the coordinates for a given area are functions of time.

Experiments were conducted using 5-in.-I.D., ¼-in.-wall, 12-in.-long cylindrical specimens. The first series of tests was made on seamless, MT-1015-20 steel, mechanical tubing. Following this series of tests, specimens made from a single centrifugal casting of type 304 stainless steel were tested. In both series of tests the cylinders were dilated hydrostatically and dynamically. The dynamic dilations were achieved through the use of SPDN 6627-6″/47 solid propellant. The propellant was contained in a right-cylindrical vented case mounted inside the vessel with its vertical centerline aligned with that of the vessel. The design of the case allowed for dispersion of the propellant gases through vents in the case wall in a rotationally symmetric pattern. The pressure-transmitting vehicle was CCl_4, which filled the annulus between the test cylinder and the propellant case.

In the first series of tests, the MT-1015-20 steel specimen ruptured at a hydrostatic pressure of ~9200 psi, and the maximum external radial dilation at midheight was 0.053 in., immediately prior to rupture. Thus the maximum elongation was ~2 per cent. Under dynamic loadings, it was found that 31 grains (114.5 g) of propellant would produce rupture, whereas 29 grains would not. It was also evident that the plastic strain prior to rupture was small.

One of the type 304 stainless steel specimens was hydrostatically dilated. At a pressure of 6700 psi, leakage occurred at the joint between the cylinder and an end closure. Rupture of the vessel did not occur, and the maximum external radial dilation corresponding to the peak pressure was 1.182 in. Following this, four additional specimens were dynamically dilated with charges of 28, 47, 72, and 100 grains, respectively, of solid propellant. The cylinder, which was dilated with 100 grains, ruptured at a pressure of ~7300 psi in 39 msec. The following tabulation gives the radial dilation as a function of the propellant charges.

Charge (grains of solid propellant)	Maximum external radial strain, in.
28	0.220
47	0.468
72	0.990
100	1.315 (prior to rupture)

The amount of strain energy absorbed as a function of distance along the axis was calculated for both the hydrostatically dilated cylinder and the one dilated with 100 grains of propellant. In the former case the strain energy absorbed at a particular axial location was ~72,500 ft/lb, whereas in the latter, it was ~84,000 ft/lb for the same strain. The increase in strain-energy absorption can be attributed to the rate of strain effect.

A comparison of the data from the two series of tests reveals that the strain-energy absorption characteristic of the type 304 stainless-

steel vessels was much greater than that of the particular MT-1015-20 specimens. However, it must be pointed out that the magnitude of the difference in energy absorption between the two steels is not only dependent on the constituents, it is greatly affected by the prior histories of the specimens. In other words, such factors as cold work and irradiation damage could have considerable influence upon the amount of strain energy absorbed by a given vessel.

In the investigation described above, pressure vessel models were tested for their ability to withstand only a rotationally symmetric internal pressure loading. However, the integrity of a reactor vessel can also be destroyed by missiles or a combination of missile and pressure loadings. Tests were conducted by Zabel of the Stanford Research Institute to investigate the penetration of plate targets by rod-like projectiles.[12,13] The first report includes an expression relating the plate thickness and the kinetic energy per unit diameter required for penetration of a mild steel plate. Since its application is limited, it is not included here. Experimental results demonstrating the effects of various parameters upon the energy of penetration are also described. The parameters were target material, distance of impact from target restraint, angle of incidence, shape of projectile, and projectile material.

Because the early experiments showed that steel plates could be penetrated more easily if they were rigidly held near the point of impact, further tests were conducted using hanging plates with free edges. Cylindrical steel rods were used as projectiles, and the projectile kinetic energy per unit diameter required to penetrate a given plate was used as a basis for comparison. The distance from the edge to the central point of impact was varied by changing the plate size. The plates were 0.120-in.-thick mild steel squares, and the sizes ranged from 6 to 24 in. on a side.

The penetration experiments showed that a critical plate size, or critical distance from the edge to the central point of impact, exists. The kinetic energy per unit diameter required for penetration when the distance is above critical is the same as that for the critical distance (see Fig. 4), and the energy is also independent of the type of support. When the distance is smaller than critical, the kinetic energy required for penetration with free boundaries increases with decreasing distance. Conversely, the energy decreases for rigid boundaries.

The important fact demonstrated by the data is the existence of a critical distance. The data given here apply only to the projectile-target system used in the experiments. Hence, additional tests should be conducted to determine the correlation of the critical distance with such parameters as plate thickness, target material, etc.

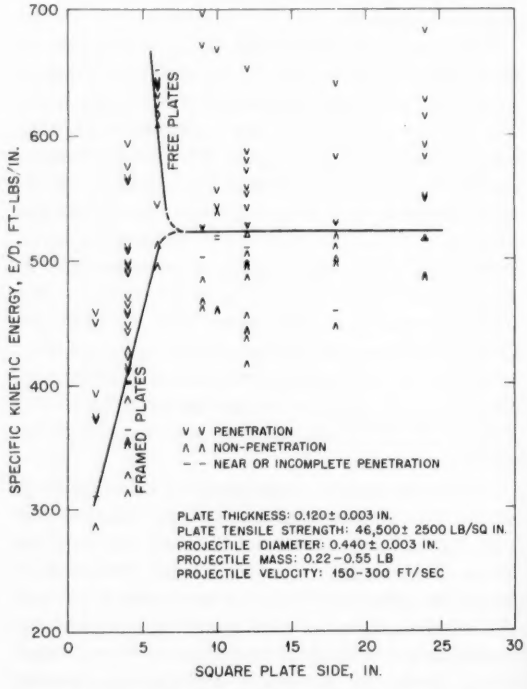

Figure 4—Threshold penetration of mild steel plates.[13]

There is evidence that the structure of the metal is important to the penetration resistance of a plate. For example, cold work or exposure to high temperatures for a long period of time could appreciably alter the resistance.

From the results obtained, the kinetic energy required for a given projectile to penetrate a given area of a pressure vessel depends both on location and the structure of the metal. Thus, a section of material with large grains which is

near a reinforcement or support would be particularly vulnerable to penetration by missiles. As suggested, such an area could be protected by mounting weakly supported baffles in the region. (B. L. Greenstreet)

Containment Vessel Leak Detection

The principal function of a containment shell is to retain the fission products from a reactor accident. In practice a maximum credible accident is defined, and the containment shell is designed so that the activity from this accident cannot leak from the containment vessel at such a rate as to subject off-site persons to excessive exposures. The leakage rate thus established is the most important criterion for the containment vessel, and the ultimate purpose of all tests and inspections of the vessel is to assure that the specified leakage rate is never exceeded. The establishment of the integrity of containment buildings of the pressure-vessel type in the United States has, in general, been based on inspections and tests of the types described below.

1. *Inspections of Weld Quality.* All shell joints of vessels built according to the ASME Boiler and Pressure Vessel Code have been fully radiographed, and shell joints of vessels built according to the API Tentative Standard 620 have been examined by spot radiography. Welds not susceptible to radiographic examination; for example, nozzles, such as air-lock penetrations, have been subjected to magnetic particle examinations. The first pass of a welded joint has also frequently been checked by a magnetic-particle examination.

2. *Spot Leak-detection Tests of Welded Seams.* Soap-bubble testing of seams has been performed both prior to and after overload pneumatic testing of vessels. In addition to the soap-bubble test, a halogen tracer-gas test involving hand probing of all welded seams was specified for the Yankee containment vessel.[14]

3. *Leakage-rate Tests Based upon Pressure-drop Observations Both before and after Equipment Is Installed.* The successful completion of the above inspections and tests is no assurance of an adequately leaktight vessel since leaks can exist in areas not inspected or tested and it is even possible to have leaks in areas that have passed x-ray inspection. Consequently, it is necessary to perform an integral leakage rate test on the vessel, even though the commonly employed pressure-drop test is rather insensitive. (The soap-bubble test, for example, will indicate a leak—if the soap is applied to the leak—that would not be detected by a pressure-drop test.)[15] Fortunately, permissible leakage rate specified for containment vessels have been of sufficient magnitude (from approximately 0.1 to 1.0 per cent contained atmospheres per day[1]) to permit testing by pressure-drop observation; otherwise leakage-rate testing would be very costly and time consuming.

Considerable difficulty is encountered in a leakage-rate testing by observation of the pressure drop because erratic data can result from changes in the temperature, relative humidity, and barometric pressure of the atmosphere in the vessel. These deleterious effects were combated in the testing of the EBWR containment vessel[16] by recirculation of the vessel's atmosphere to prevent stratification and by monitoring vessel and atmosphere temperatures, the relative humidity, and the barometric pressure in an effort to detect obviously spurious readings. Even with this careful approach, eight days of testing were necessary to establish accurately the trend of the leakage-rate curve.

Temperature compensation was provided in the leakage-rate testing of the General Electric Vallecitos Boiling Water Reactor Plant, the Wright-Patterson Air Field Plant, and the Dresden Nuclear Power Station containment vessels.[17,18] Small preleak-tested chambers were interspersed throughout the containment vessels and interconnected with tubing to one leg of a manometer which had its other leg connected to the containment vessel atmosphere. Although there were slight temperature lags between chambers and containment vessels,[17] temperature effects on leakage-rate detection were notably reduced.

Since air locks and other penetrations of the containment shell are critical components of the containment shell and perhaps more susceptible to leakage than the containment shell wall, particular care has been given to the leak

testing of these items by soap-bubble testing[16,17,19] to ensure that all seals are tight.

Continued assurance of integrity is a problem that has been neglected to a large extent. However, the Yankee containment vessel will be continuously monitored by maintaining a slight, positive pressure by means of make-up from a manifold of compressed-air cylinders and by monitoring the weight loss over a period of time. At Dresden periodic soap-bubble leak-detection tests at 2 psig are planned for critical seals of the containment vessel, such as air-locks and bulkhead penetrations. Provisions such as have been made at these installations are indicative of the present trend toward providing adequate containment not only at the completion of construction but also throughout the life of the plant. . (G. C. Robinson)

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* Second United Nations International Conference on the Peaceful Uses of Atomic Energy, Geneva, September 1958. (Individual papers are available from the Office of Technical Services, Dept. of Commerce, Washington 25, D. C. Bound volumes of papers are available from United Nations, Office of Public Information, New York.)

Activity Build-up

The highly radioactive coolant of a liquid-fueled reactor system has always posed a serious design problem. The problem is reduced in solid fuel reactors by placing two barriers between the fission products and the coolant: (1) the solid fuel itself and (2) the canning or cladding material. Coolant activity in solid fuel reactors is largely the result of corrosion product activation. While low compared to activity levels encountered in fluid fuel reactors, these levels are still high enough to require elaborate decontamination measures.

Experimental data are being obtained on the build-up of long-lived activity (half life greater than about 50 days) in stainless-steel-clad water-cooled reactor systems at the Army Package Power Reactors (SM-1, formerly designated APPR-1), Fort Belvoir, Va.[1–3] The information obtained from the SM-1 has been summarized in a review article.[4] Some of the significant findings of this work are presented here:

1. During the first year of SM-1 operation, coolant activity has been dominated by Co^{58} activity (71-day half life) resulting from the reaction $Ni^{58}(n,p)Co^{58}$.
2. The Co^{60} activity (5.2-year half life) resulting from neutron captures in the stainless-steel impurity Co^{59} is significant and will ultimately reach and pass the Co^{58} activity (assuming that the present trend continues).
3. Coolant temperature has a marked effect on surface deposition of coolant activity. More activity is deposited on samples in high-temperature water than on samples in low-temperature water.

The importance of the $Ni^{58}(n,p)Co^{58}$ reaction was unexpected because of a lack of knowledge of the cross section of this high-energy reaction (the Coulomb barrier for the proton is about 5 Mev). The appearance of the Co^{60} activity was also unexpected because the Co^{59} impurity was of negligible concern to previous users of stainless steel and consequently was not included in the producer's analysis. This revelation has influenced the Co^{59} specification for Core II of the SM-1 reactor,[5] as well as the specifications for the reactor core materials to be used in the N. S. *Savannah*.[6,7]

Other major contributors to the build-up of long-lived activity in the SM-1 primary system are Fe^{59}, Mn^{54}, and Cr^{51}, all of which are components of type 304 stainless steel. A research and development program has been initiated to determine ways of decontaminating the SM-1 primary system. Preliminary results[8] indicate that a caustic permanganate treatment followed by an acid rinse may achieve decontamination factors of about 20 without serious attack of the stainless steel.

The operation of the Shippingport Pressurized Water Reactor (PWR) should provide useful data on the build-up of activity in the water coolant of a reactor with zirconium-clad fuel. An extensive program[9] is being initiated to measure the radiation field around the PWR as a function of time. An important part of this program is the so-called "hair-pin loop" facility which contains a representative sample of the circulating water coolant and which can readily be isolated from the main system for detailed examination. The hair-pin loop is to be monitored periodically with a scintillation counter which can be coupled with a gamma-ray spectrometer, if necessary. About every six months the hair-pin loop is to be disassembled and radiochemically analyzed for fission products, as well as for hafnium, iron, and cobalt activities. Since the PWR core contains very little stainless steel, the build-up of cobalt activity in the water coolant should provide information on the importance (with respect to coolant activity build-up) of cobalt-bearing materials outside the core but within the primary system (pressure vessel, piping, etc.). Data from the hair-pin loop facility are to be correlated with periodic activity surveys of expected primary system hot spots (pumps, valves, boiler inlet and outlet, etc.), as well as with information from the

failed fuel element detection system and with the monitoring results from the crud probe.

Operation of the SM-1 and PWR reactors should provide very useful data on primary system activity build-up for the reactor designer. However, before the build-up of activity as a function of space and time can be predicted, much more basic research will be needed to determine how activation products are formed and where they are deposited. (E. E. Gross)

Fission-product Release

An investigation of a simulated nuclear accident involving melting of a full-size zirconium-alloy PWR fuel subassembly has been conducted by the Mine Safety Appliance Research Corp. (MSA) in cooperation with the Westinghouse Atomic Power Division.[10] The experiment was carried out to determine the magnitude of fission-product release and the resulting distribution of the fission products by plating, atmospheric release, and subsequent fallout. The fuel assembly used consisted of nine Zircaloy-clad plates with the central plate removable. The assembly was irradiated at a very low integrated flux of 2×10^{14} neutrons/cm^2 in the Battelle reactor. The $2\frac{1}{2}$- by $1\frac{5}{8}$- by 15-in. assembly was mounted with the bottom half inserted in the working coil of an induction heater. Melting was carried out in a steam atmosphere, with the melt being allowed to drop into water.

The experimental apparatus consisted of a thick-walled stainless-steel meltdown chamber of about 50-gal capacity, a "fall-out" chamber of about 100-gal capacity, and a steam generator and condenser. A charcoal trap was used for collection of active gases. To obtain analytical material balances, a small punched sample, irradiated simultaneously with the fuel bearing plate, was put into solution as a standard. Low counting rates might have affected the final results to some extent.

The results of previous work at ORNL[11] on a miniature scale were available as a guide in this investigation.

The following conclusions summarize the principal data presented in the paper.

1. Of the metal actually melted, about 20 per cent reacted with the water and hydrogen was liberated, but there was no self-sustaining reaction and no apparent effect on the fission-product release.

2. The molten metal completely filled the voids in the assembly and was observed to stream rather than drip from the channels.

3. Release of xenon ranged from 42 to 58 per cent and was relatively independent of meltdown characteristics. (This observation is inconsistent with the ORNL results, which indicated 100 per cent release.) The quantity of xenon released has been interpreted as a measure of the percentage of core metal actually melted.[12]

4. Iodine was released in amounts of 1.3 to 7.6 per cent and from 7.9 to 25.8 per cent under conditions of low and high heat inputs, respectively.

5. Strontium and barium released from the fuel ranged from 0.7 to 1.6 per cent. About 10 percent of this was found beyond the meltdown chamber.

6. About 0.6 per cent of the cerium and yttrium was released, with little more than one-tenth being outside the initial chamber.

7. Cesium release ranged from 0.4 to 0.5 per cent. This apparent suppression of cesium release is attributed to its behavior at trace concentration.

A recent ORNL report describes an extensive series of experiments in melting irradiated reactor fuels under conditions intended to simulate the loss-of-coolant type of accident.[13] Particular emphasis has been placed on the APPR stainless-steel-clad UO_2 fuel and on zirconium-clad uranium-zirconium alloy fuel plates of the STR type. Other fuels examined, some only superficially, were MTR type aluminum alloy fuels, Geneva type aluminum-U_3O_8 dispersion fuels, solid UO_2 fuels, and the GE-HTRE fuel.

The status of work of Brewer[14] on the pyrometallurgical spent-fuel-recycle program and on high-temperature thermodynamics is cited as a guide to be used in anticipating logical behavior of the fission elements. Since even the simplest concept of a loss-of-coolant accident cannot avoid including a number of variables, some attention has been given to the effects of time of heating, total time molten, presence or absence of an oxidizing atmosphere, presence of exposed or concealed edge of fuel, and heating to temperatures below the melting point. The special effect of burn-up is considered separately.

In a typical experiment, a small specimen or miniature fuel plate was melted. This is usually done by induction heating. Volatile components were selectively trapped and radiochemically

identified. Material balances were developed to indicate percentages of each fission product either contained or dispersed from the source.

There has been relatively little experimental work of a similar nature reported,[15] and therefore it was of special interest to compare these results with those of the somewhat larger scale work conducted by MSA.[10] Table V-1 is a comparison of the ORNL results of the fission-product release from Zircaloy-clad STR fuel plates with the MSA results for Zircaloy-clad plates.

Table V-1 FISSION-PRODUCT VOLATILIZATION BY MELTING ENCAPSULATED ZIRCALOY PUNCHED DISKS AFTER 15 PER CENT BURN-UP

	ORNL results[13]		
	For melting in air	For melting in steam	MSA* results for melting in steam
Heating time, sec	20	30	
Activity released (% of total present, averaged results of several tests):			
Rare gases	100	100	42–58
Gross gamma	3.4	6.3	
Iodine	26.2	46.6	7.9–25.8
Cesium	10.6	18.7	0.5
Strontium	0.85	1.8	0.7–1.3
Barium	0.2	0.4	0.5–1.6
Cerium	0.004	0.03	0.6

*Mine Safety Appliance Research Corp.

The consistently lower release values observed by MSA for all fission products other than cesium are believed to be adequately explained as an apparent measure of the percentage of the fuel plate actually melted. Their very low cesium value is believed to be an outstanding example of the burn-up effect since the MSA results were obtained for trace concentrations of fission products.

The following conclusions seem to be justified by the accumulated data.

1. The composition of the fission products released to the atmosphere from a quiet meltdown of reactor fuel is limited mainly to the "volatile constituents" whose boiling points are less than the melting temperature (or the maximum temperature obtained during ignition) of the fuel.

2. The release values are affected by length of time of heating, the concentration of fission products (burn-up), degree of oxidation, and type of fuel, etc.

3. All results indicate an exponential correlation with the fuel melting temperature when other conditions are held constant. (G. W. Parker)

Environmental Monitoring

During the period immediately following a radiation incident involving environmental contamination, much of the monitoring activity is directed toward the accumulation of data with which the effects of the contamination are measured. In many cases the ultimate value of the data is unknown at the time of collection, but, in view of a possible uncertainty in what and how much monitoring data are needed, it is generally advisable to obtain as much as possible under the circumstances prevailing at the time. In addition, the disappearance of the contamination through physical and biological processes may allow little time in which to plan a detailed monitoring program *subsequent* to an accident. Preplanning can be of considerable value, but this could easily be obviated by the nature of the accident.

In the process of data interpretation, the problem of correlating the various pieces of information can be troublesome. An interesting account of some of the work carried out by the Atomic Energy Research Establishment (AERE) following the Windscale accident has been reported.[16] The report describes the measurements of the I^{131} activity in samples of soil, grass, and milk, as well as dose rates found in the contaminated areas. The correlation between these measurements is also discussed in considerable detail.

Scintillation counters were used to determine the dose rates in the areas contaminated by the accident. The measured dose rates were then converted into equivalent surface contamination in units of microcuries per square meter by adjusting the theoretical dose rate above infinite plane sources uniformly contaminated with various gamma-emitting isotopes with the known energy response characteristics of the instrument. This theoretical calibration was checked by measuring the dose rate above an array of discrete sources of the same three types (Cs^{137}, I^{131}, $Te^{132} + I^{132}$) as those used in calculating the dose rates. After making a correction for the finite size of the array, the agreement for the two methods was good. It was found that the type of surface upon which the array was located had a measurable effect on the dose rates, but this

would be difficult to allow for in the calculations and probably is not justified in view of the good agreement.

Dose rates are reported for a seven-week period beginning the day after the accident for a point near Seascale, about 1.7 miles from the site of the accident. The background dose rate was obtained eight months following the accident and was used to correct the raw monitoring data. The resulting decay curve fits the decay of 8d I^{131} quite well during the last three weeks of the reported data. Another component of the gross decay curve was resolved by subtracting the I^{131} extrapolated contribution during the first four weeks; this was assumed to be 3.2d $Te^{132} + I^{132}$. Table V-2 gives the ground contamination and the dose rates of 1 m due to the two assumed contributors, as well as the values for Cs^{137} obtained from laboratory measurements on the soil.

Table V-2 GROUND CONTAMINATION AND 1-M DOSE RATES AT SEASCALE ON OCT. 11, 1957

Isotopes	1-m dose rate, mr/hr	Ground contamination, $\mu c/m^2$
I^{131}	0.12	17
$Te^{132} + I^{132}$	0.37	10
Cs^{137}	0.003	0.3

The activity ratio, I^{131}/Te^{132}, in the undistributed fission products on Oct. 11, 1957, was calculated to be 1 : 0.75. This agrees remarkably well with the measured value of 1 : 0.6, as shown in Table V-2. By assuming this ratio to be constant throughout the fall-out zone, the ground contamination at other locations was determined.

Samples of grass and soil were obtained from various areas and taken to the laboratory for more precise measurements by gamma spectroscopy. Excellent agreement was obtained with the two independent methods. The laboratory measurements also provided the only estimate of the Cs^{137} levels.

Aerial surveys were made on the ninth and twelfth days following the accident with gamma scintillation counters flown at an altitude of 500 ft. Uncertainties in the background of the contaminated area limited this interesting technique to those areas where the counting rate was at least five times the background rate. The background rate was estimated from flights over water and over uncontaminated ground of known geology. Unfortunately, in only a few locations did the position of the aircraft correspond with the location of one of the ground measurements. Even so, in the five cases reported, the ratio of the counting rate to the dose measured at 1 m was reasonably constant. With the correlations observed, a counting rate at 500 ft equal to twice the background would correspond to a 1-m dose rate from these same contaminants of about 30 $\mu r/hr$.

The most interesting aspect of this paper deals with the relationship of the I^{131} activity in milk from the Seascale area and the ground contamination previously mentioned. The observed ratio of the I^{131} ground contamination to the milk contamination increased by a factor of about 5 during the period of the measurements. This was explained as resulting from a transference of the activity from the grass to the ground by both physical and biological processes (leaching by rain and translocation to the grass roots). Of course, field measurements of the ground contamination cannot distinguish between I^{131} in the soil and that in or on the grass. The ratio varied from 17 $\mu c/m^2$ per μc/liter on October 12 to about 95 $\mu c/m^2$ per μc/liter on November 28. With the survey instruments used it was estimated that the minimum level of ground contamination which could be reliably measured was 0.24 $\mu c/m^2$. This corresponds to a milk activity of 0.014 μc/liter at short times following such an accident.

Additional results of the laboratory measurements, as well as the details of the analytical techniques, are reported in a companion paper.[17] Small thallium-activated sodium iodide crystals were used with single-channel analyzers for relatively pure samples, while multichannel analyzers were used when several gamma emitters were present in the sample. Quoting from reference 17:

"In the first few days after the incident, I^{131} activity in the samples was predominant to such an extent as to mask other activities. By retaining samples until the I^{131} had decayed, the presence of other nuclides was demonstrated. The fission products Cs^{137}, Ru^{103}, Ru^{106}, and Zr^{95}, with their daughter products, were found in the herbage and soil, and Cs^{137} was found in the milk.

"Although the dissemination of I^{131} from the reactor reached a peak in the early hours of October 11 and ceased by about noon on that day, the activity did not appear in the milk of local cows until the afternoon of October 11. The maximum activity in the milk was reached on

the afternoon of October 12 or the morning of October 13."

The I^{131} activity in samples (from remote areas) of milk ranged from less than 0.3 mμc/liter to well over 1000 mμc/liter for a few samples from points close to Windscale. An average value for the apparent half life of the I^{131} in the milk was found to be five days. Results through October 28 were corrected back to October 11, using the five-day half life, and were plotted on maps of the area. The picture of fall-out distribution agreed with what might be expected from the meteorological conditions and with observations on air filters from different parts of England.

Samples were analyzed for Cs^{137} after sufficient time had elapsed for the I^{131} to decay to an insignificant level. The maximum Cs^{137} activity in milk was found to be 15 mμc/liter in a sample taken about 8 miles from the plant.

Many herbage samples were collected from the area near Windscale and from various areas up to 200 miles from the stack. Samples analyzed for I^{131} ranged in activity from 0.2 to 82 $\mu c/m^2$ and again exhibited an apparent half life of 5 days. Results of the analyses, when plotted on a map, gave contours which agreed well with those obtained from the map of the I^{131} activity in milk. After decay of the I^{131} activity to a negligible level, the following gamma-emitting isotopes were identified: Ce^{141}, Ce^{144}, Ru^{103}, Ru^{106}-Rh^{106}, Cs^{137}, and Zr^{95}-Nb^{95}. Quoting further from reference 17:

"Soil samples were taken at a number of points in the fall-out area from which herbage samples were taken. Analyses were made for Sr^{90}, as well as for the gamma-emitting isotopes. The low proportion of Sr^{90} in the herbage relative to that in the soil indicated that most of the Sr^{90} in the soil was not associated with the accident.

"When the results for I^{131} in herbage were compared with the level of activity in milk from the same area, a correlation was obtained between the herbage activity and the milk activity. The median of 30 ratios showed that 11 $\mu c/m^2$ of herbage produced 1 μc/liter of milk. . . . An estimate of the total deposition of I^{131} was made from the herbage contour maps, giving a figure of about 20,000 curies, of which about 11,000 curies were deposited in Great Britain. . . .

"The versatility of the gamma-scintillation spectrometer was demonstrated by the measurement of the gamma activity of numerous biological samples in many different forms and of widely varying specific activity. One chief advantage which this method holds over conventional chemical separations is the small amount of preparation required for samples."

(H. H. Abee)

Environmental monitoring of the area surrounding a nuclear facility during its normal and routine operation has also been considered. In general, such monitoring programs differ from those previously described only in the level of activity found. The routine monitoring program is usually concerned with many samples of low activity, and, almost without exception, laboratory techniques are required to obtain valid analyses.

While the pro's and con's of routine environmental monitoring programs are debated at intervals, almost everyone agrees to at least a minimal effort. The exceptions are those who feel that in the case of power reactors environmental surveys are not necessary except in cases of accidental releases,[18] the argument being that routine escape of activity is negligibly small and, in any event, can be more easily monitored at the source. The preponderance of opinion, however, seems opposed to this view. The recent reductions in the maximum permissible exposures will have some effect on this very point.

One proponent for the complete monitoring program has recently published his views.[19] His paper provides a descriptive summary of the various pre- and post-startup surveys he recommends for a power reactor facility. The preoperational surveys would continue for a period of 12 months or more to provide seasonal coverage. The experience and information acquired from this program would then be used in the design of the operational survey program. While various environmental factors, as well as the specific reactor type, may modify the outlined survey, the recommended program consists of routine field surveys for activity in samples of air, water, and soil. For the most part the recommendations will be familiar to those with experience in this field.

The air-monitoring program is limited to consideration of airborne radioactive particulates, except for a casual reference to A^{41} measurements, which are said to be unnecessary in the case of water or sodium-cooled reactors. (This latter statement would appear to be questionable in view of the experience of the Georgia Nuclear Laboratories with the Radiation Effects Reactor (RER) located at Marietta.) No mention is made

of the value of autoradiographs obtained from air-filter samples and gummed fall-out collectors.

The discussion of water-monitoring requirements is much more detailed. Ground-water and surface-water monitoring are discussed, as well as the advisability of the sampling of water biota (algae, fungi, and plankton) and silt and mud from the stream bottoms. It is indicated that concentration mechanisms are important in the uptake of the different radioisotopes and that this effect must be allowed for in the sampling program. For example, in fish, strontium and cesium might best be located in bone and muscle, respectively. Not indicated, but certainly obvious, is the fact that an expert in this field will be required to make the interpretation of the data collected from such a comprehensive program. The sampling program for radioactivity in the soil, vegetation, and in land animals is also quite comprehensive, but, again, the recommendations are purely descriptive.

For the sampling programs described it is estimated that the capital cost would be $55,500. Of this amount about $35,000 represents the cost of five monitoring stations, each equipped with automatic continuous counting equipment for the air-monitoring program. The remainder of the estimated captial cost is for laboratory facilities to handle the other samples.

The legal significance of original records is pointed out; film records are suggested to minimize storage problems. The personnel required for the described programs are estimated at four, three of whom would be technicians.

(R. L. Clark)

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Nuclear Incidents

As used in this Review "nuclear incident" is defined as a nuclear accident in which one of the following occurred: (1) serious personnel exposure, (2) significant contamination of the surroundings, or (3) substantial damage to the reactor. Two incidents reported here, the Los Alamos accident and the Boris Kidrich Institute of Nuclear Studies (Yugoslavia) accident, resulted from inadvertent critical assemblies. The Los Alamos accident occurred in chemical processing equipment, and the Kidrich accident was in a critical experiments facility. In both instances the nuclear excursion resulted in a radiation fatality.

Analyses of both of these incidents indicate that they could have been avoided. In the case of the Los Alamos accident, which was directly attributable to operator errors, changes in equipment and procedure are planned to reduce the probability of the recurrence of such an accident. The low incidence of radiation injuries in American nuclear plants and laboratories is largely attributable to the safety philosophy and constant vigilance of the Atomic Energy Commission. The possible consequence of an environment where such control is not exercised is illustrated in the various descriptions of the Kidrich Institute accident.

Los Alamos Criticality Accident

The early reports[1,2] on the Los Alamos criticality accident of Dec. 30, 1958, covered only the bare facts on the accident and stated that the operator, Cecil W. Kelly, died of radiation damage to the central nervous system. The results of the official investigation of the accident were reported by H. C. Paxton and others.[3,4]

At the time of the accident, a physical inventory was in progress in the TBP solvent extraction plant used to recover and concentrate "lean" residues from a plutonium recovery operation. Residual materials in all process vessels were to be evaluated for plutonium content.

Reconstruction of significant events leading up to the accident indicates that plutonium-rich solids, which normally would have been handled separately, were washed from two other vessels into a vessel that contained dilute aqueous and organic solutions. After removal of most of the aqueous solution, the remaining material was transferred to the solvent-treating tank in which the accident occurred. In addition, concentrated nitric acid was used to wash solids from the bottom of the vessel into the solvent-treating tank. The solvent-treating tank already contained a caustic-stabilized aqueous-organic emulsion. Transfers were followed by air-sparging, and therefore the emulsion was broken by acidification, the plutonium originally in the solids was extracted into the solvent, and the phases separated.

The resulting mixture in the 225-gal, 38-in.-diameter, solvent-treating tank immediately before the accident is believed to have consisted of 87.4 gal of aqueous solution containing 40 g of plutonium and, on top, a 42.2-gal layer of solvent containing 3.27 kg of plutonium. Solids (containing 60 g of plutonium) were suspended in both the aqueous and organic phases and at the interface. Estimates indicate that the 8-in.-thick solvent layer was barely (roughly 5 dollars) subcritical at the plutonium concentration of 20 g/liter.

A chemical operator started the motor for driving the stirrer in the solvent-treating tank. The initial action of the stirrer in the solvent-treating tank forced aqueous solution up along the wall, displacing the outer portion of the solvent layer and thickening the central portion. There was a "blue flash" and a muffled report. The burst was certainly terminated by the violent disturbance that it generated, and continuation of the stirring diluted the plutonium beyond the point at which a critical reaction could recur.

The operator, who was looking into a sight glass at the top of the tank, was knocked off

the two-step ladder on which he stood. He apparently turned the stirrer motor off and then on again, ran out a nearby exterior door, and called that he was "burning up." Although the shock displaced the tank about $\frac{3}{8}$ in. at its supports, it was not ruptured and no plutonium escaped.

At the time of the burst, a second operator, who was about 40 ft away in an adjoining room, saw a reflection of the light on the walls, heard the report, went to help, and was joined by another operator. These two operators led the injured operator to a shower. As the three passed the tank in which the accident occurred, the second operator turned off the stirrer.

The injured operator went into deep shock within 15 min of exposure. He regained consciousness about 6 hr later and remained rational and confortable until nearly the time of his death, about 35 hr after exposure. The operators who went to his assistance showed no physical effect of exposure other than a typical change in blood count in the case of the man who arrived first and switched off the stirrer motor. Both have continued on regular duty. The fatally injured man received a radiation dose estimated at 12,000 rem (±50 per cent). The next highest doses received by any of the operating personnel were estimated at 134 and 53 rem. Other exposures from the supercritical burst were less than 4 rem. Radiochemical analyses for Mo^{99} indicate that the total number of fissions in the burst was about 1.5×10^{17}.

A nuclear-criticality review of the entire plutonium plant had been completed about a month before the accident. Changes in equipment were planned to reduce dependence on procedural control of batch sizes for nuclear safety purposes. The existing type of control had been used successfully during the $7\frac{1}{2}$ years that the plant has been in operation.

Since the accident there has been further detailed review of the entire plutonium plant, and the following additional safety steps have been taken:

1. Geometrically favorable dissolvers and feed tanks have been designed for filtration and solvent extraction.

2. Supplementary transfer lines have been blocked to minimize the opportunity for abnormal interchanges.

3. Cadmium nitrate solution has been placed in vent tanks and vacuum-buffer tanks to protect against criticality in case plutonium solutions are accidentally introduced. The use of both soluble and fixed neutron poisons in large process equipment is being studied. Both these methods present difficulties since the soluble poison may result in unequal distribution between the phases and there is a possibility of plutonium deposition on the surface of the fixed poison.

4. Neutron detectors for indicating abnormal deposits of plutonium have been calibrated and tested.

The investigation review committee found[5,6] that the accident was directly attributable to errors on the part of the deceased operator in handling several batches of material together instead of one at a time. They also found that the procedures for this process were such that safety of operation depended substantially on the ability and judgment of individual operators but that the incident might have been prevented had the organizational arrangement required closer supervision to ensure that normal procedures were followed. The Committee commended the Los Alamos Scientific Laboratory on the thorough evaluation it made of the accident and on the planning of measures to be taken to prevent a recurrence.

(C. E. Guthrie)

Kidrich Institute Accident

The natural-uranium-fueled, D_2O-moderated, unreflected, critical assembly at the Boris Kidrich Institute of Nuclear Science, Yugoslavia, became critical inadvertently on Oct. 15, 1958, and caused significant radiation exposure to eight members of the technical staff. One fatality resulted. The incident has been recently described by Savić;[7] an earlier report appeared in *Nucleonics*.[8]

The core of the assembly[9] is a lattice of natural uranium rods, each 210 cm long and 2.5 cm in diameter, located at a 12-cm center-to-center spacing. The over-all lattice diameter is about 200 cm, and the uranium inventory is about 4 tons. Control is primarily effected by adjustment of the heavy-water level as it is transferred, by a pump, from a storage tank located below floor level. Water can be pumped into the tank at two rates corresponding, at

near critical, to 3.75 and 1.2 per cent δk/min. Neutron detection instrumentation is conventional, consisting of three BF_3 proportional counters with associated scalers and recorders. There is at least one logarithmic rate meter.

The safety system consists of a "control key" through which power is apparently supplied to two magnetically supported, gravity actuated cadmium safety rods, each reportedly being sufficient to hold the system subcritical separately even with the tank completely filled with D_2O. Dose rate meters for both thermal and fast neutrons and for gamma radiation give audible signals of radiation fields above preset rate levels. Suitable circuitry is provided to shut down the reactor, by dropping the rods, upon radiation signal. It was apparently possible to start the reactor in a normal manner with the instrumentation de-energized. (The relation of the magnet supply at the time of the accident to this condition is not clear, if the worth of the rods is that reported.) The assembly is unshielded.

At the time of the accident the personnel are reported to have been engaged in a "subcritical experiment" at distances of 3 to 8 meters from the tank. No monitor, safety, or control instrumentation was operating. By a means which is not mentioned, D_2O was transferred (possibly pumped) from the storage tank into the assembly tank, through a vertical distance of at least 4 meters, without cognizance by any of the persons present. The resulting supercritical assembly was first detected by an odor of ozone. The immediate actions, which no doubt stopped the reaction, are not described. The time of the energy release is indicated to be the order of 10 min from a record by an air monitor some 540 meters distant, and the magnitude was evaluated as 8×10^7 watt-sec (2.4×10^{18} fissions). The power versus time pattern is not reported and, presumably, not determinable.

Personnel exposures to thermal neutrons were estimated from activities induced in metal objects carried by the individuals and in body fluids. Epithermal-neutron (<1 Mev) and fast-neutron exposures and the gamma-ray exposure were estimated from the characteristics of the reactor core and from measurements of the leakage spectrum. The estimated average whole-body dose was 683 rem, which was made up of 49 rem from thermal neutrons, 223 and 116 from epithermal and fast neutrons, respectively, and 295 rem from gamma rays. Body areas nearer the points of incidence of the neutrons experienced, of course, exposures estimated to be two or three times the whole-body average.

The six most severely exposed patients were treated at the Curie Foundation in Paris.[9] Five of the patients received bone marrow treatment (~10 billion nucleated cells of aspirated marrow from healthy donors). The patient who received the greatest exposure died a month after the accident from complications in the respiratory and gastrointestinal tracts. The other four patients as well as the sixth worker who did not require bone marrow treatment are recovering.

Lack of a detailed description of the actual cause of the accident and, to a lesser extent, of the manner in which the reaction was stopped preclude any well-defined conclusions being drawn for the guidance of others operating in this or related fields. It is quite obvious that there was a serious design deficiency in the electromechanical interlocks intended to prohibit unattended and uncontrolled operation and/or a significant failure in their functioning, provided, of course, the operational philosophy of the Institute demands such interlocks. If, as an alternate, more reliance is placed on administrative control, it, too, was not thorough. To be more definitive in an analysis would require detailed knowledge of several special situations during the incident, including the location of the control key and of the safety rods, the operation of the electrical circuits controlled by the key and of the circuit supplying the safety-rod magnets, and other items that are individually of small import but which in combination give insight into the sequence of events leading to the supercritical assembly.

A mild sense of complacency can be detected in the report[10] of data on the leakage radiation when the assembly is critical at low power. These data are purported to show that "this reactor, although completely unprotected, could have been used safely up to the critical point and even over critical with a period of a few minutes for the personnel around the reactor." This analysis may have been corrected so long as an unexpected and unplanned event, such as

that being reviewed, did not occur. The analysis led to a false sense of security. Since none of the nuclear accidents which have occurred, either in critical experiment facilities or in chemical process operations, were the result of an "expected" procedure, personnel protection must always be provided for the maximum "unexpected" event. It is well known that although, in general, the consequences of a nuclear accident in a critical assembly will have a much more limited range than those of a reactor runaway, the very nature of critical assemblies and their operation makes them more susceptible to accidents. Therefore equal protection of personnel nearby the two types of operation is necessary. The inclusion of shielding materials or of protection-by-distance is general practice in the design and use of critical experiment laboratories even for "zero" power operations. Lack of detailed knowledge of the operation leading to this accident precludes drawing any conclusions regarding the relation of this tragic occurrence and the philosophy implied in the above quotation. The concluding statement of one of the papers reports the construction of an underground control room "with which even higher excursions with the reactor might be made." It is not clear that such caution is exercised in all critical experiments. (A. D. Callihan)

*Regulatory Role of the Atomic Energy Commission**

The Atomic Energy Act of 1954 requires that licenses be granted by the Atomic Energy Commission before nuclear facilities can be built and operated by institutions and organizations outside the Federal Government's program. Before such licenses can be given, the Commission requires reasonable assurance that the health and safety of the public will not be unduly endangered by operation of the proposed facilities. It is the responsibility of the applicant to design a facility for which safe operation can be expected, and to present fully his plans for construction and operation, for independent analysis and evaluation by the Commission. The Commission in turn is responsible not only for evaluating the safety of proposed facilities and making periodic inspections of their operation, but also has the obligation to make all information accumulated from its research and development programs fully accessible to any parties interested in designing reactors.

Thus, the present procedures and regulations followed in arriving at decisions on the safety of proposed facilities may be summarized as follows:

1. The Atomic Energy Commission makes available to potential reactor designers and other interested parties information pertaining to the safety of nuclear facilities.
2. An applicant is required to present full information to the Commission on his proposed plans, details of the reactor design, analysis of probabilities and consequences of possible accidents, and the evidence for his belief that the proposed system can be operated safely.
3. The Division of Licensing and Regulation of the Commission makes analysis and evaluation of the potential hazards of the facility and the adequacy of safeguards provided.
4. If the facility is authorized, the Inspection Division of the Commission determines the state of compliance with design criteria and specifications and evaluates safety of operation as described below.

Factors Considered in Reactor Site Evaluation

Before a license for operation can be given, the basic decision required is that there is reasonable assurance that the facility can be constructed and operated at the proposed location without undue risk to the health and safety of the public. Two essential criteria that are applied to any proposed site-reactor combination in arriving at this decision are: (1) in normal effluents of plant operation, the radioactivity released must not result in levels beyond the site boundary in excess of the maximum per-

*This section is based substantially on Geneva Conference paper, A/CONF.15/P/2407, written by Dr. C. K. Beck, Licensing and Regulation Division, U. S. Atomic Energy Commission, Drs. M. M. Mann and P. A. Morris, Inspection Division, U. S. Atomic Energy Commission.

missible levels for continuous exposures; and (2) from possible accidents, the radioactivity that might be released, even from the worst accident whose occurrence is considered credible and under the most pessimistic dispersion conditions, must not result in doses beyond the site boundary in excess of permissible (once in a lifetime) emergency doses.

The important features of the proposed site must be considered in applying these criteria. But the decision cannot be made on site data alone. It is equally important that the features of the reactor and of its associated equipment be considered. Thus, the reactor and the site must be evaluated together, and it follows that certain safety deficiencies or inadequacies in either the reactor or the site can be offset in part by compensating characteristics in the other.

The technical safety analysis and evaluation of any proposed reactor-site combination is made largely on the basis of information presented by the applicant in a Hazards Summary Report. In practice, the report usually consists of two parts: (1) a Preliminary Hazards Report, presented relatively early in the life of a project, when a construction permit is sought, and when the design features of the plant are still far from crystallized, and (2) a Final Hazards Report, presented when the design is completed, construction is nearly complete, and an operating license is being sought.

These reports together present several categories of information, from which the necessary decisions on safety, as indicated above, may be made.

1. *Description of pertinent site data:* The applicant here gives a complete quantitative description of all environmental features relative to safety (or hazards) of the facility, the size and location of the applicant's property holding, the position of the facility on the property, population and its distribution in the surrounding areas, a summary of important features of meteorology, hydrology, geological characteristics, and seismological history. These data are summarized by the applicant with an identification of the significant features, favorable or unfavorable with respect to the safety aspects of the proposed facility.

2. *Description of the reactor:* This includes description of the nuclear, thermal, hydraulic, mechanical and chemical characteristics, shielding features, and instrumentation and control systems. These descriptions are related to the functions to be performed and to the basis for the final choice of performance specifications.

3. *Description of auxiliary systems:* The design of the systems is delineated, with emphasis on the features relevant to safety. Dimensions, functional arrangement of the building, containment specifications, waste disposal systems, power and cooling systems, and the potential interaction of these features with the reactor are described.

4. *Outline of administration and organization, operating plans, policies, and procedures:* Description is given of the organizational structure, the areas of authority and responsibility of the operating staff, the procedural ground rules and policies, and the plans and procedures to be followed in case of accident or emergencies.

5. *Analysis of the extent of hazards and the adequacy of safeguards:* Potential hazards that could arise from operation of the facility are defined. Description leads to a set of assumptions, realistically related to the characteristics of the reactor and the design of the facility, which would result in an accident having consequent damages not expected to be exceeded by any other credible accident. Having defined these assumptions, analysis is made of the consequences both to the occupants of the facility and the inhabitants of surrounding public areas, and it is required that the consequent damages to the public, even from this maximum credible accident, would not be beyond those considered acceptable.

Primary staff responsibility for reactor hazards evaluation rests with the Hazards Evaluation Branch of the Division of Licensing and Regulation. The Commission is assisted in carrying out its reactor safety functions by the Advisory Committee on Reactor Safeguards. This statutory committee reviews safety studies and facility licensing applications referred to it and makes reports thereon, advises the Commission with regard to the hazard of proposed or existing reactor facilities and the adequacy of proposed reactor safety standards, and performs such other duties as the Commission may request. The Committee's report on application for facility licenses becomes a part of the rec-

ord application and is available to the public, except for security material. Members are appointed by the Commission for a term of four years, and one member is designated by the Commission as its chairman.

Inspection Objectives and Criteria

Objectives

As stated previously, safety of reactor operation is influenced by two principal factors: (1) the design of the reactor and its associated equipment, and (2) the manner in which the reactor is operated.

The design of a reactor and the manner in which it is constructed define a number of complex relationships among the physical characteristics of the reactor, its location and containment, the instrumentation, and the control and safety systems. These relationships are evaluated, to a large extent, before an operating license is issued, by the Division of Licensing and Regulation as part of its over-all review of the proposed design and operating plans. The evaluation of these relationships, by itself, serves to define the maximum credible accident and, therefore, the maximum potential hazard to the public.

Whether or not the reactor can be operated safely depends not only on design features and inherent safeguards, but also upon the operational procedures, the training and skill of people, and the effectiveness of administrative control and management. The effect of these factors on safety of reactor operation can be evaluated only partially during construction of the reactor. Final evaluation can be made only when the reactor is completely constructed and is operated according to established procedures, by the normal operating organization. This evaluation made by the Division of Inspection, together with the evaluation of the physical characteristics, serves to define the probability that a serious reactor incident may occur.

Objectives of reactor inspection are stated as follows:

1. To determine that the facility is actually constructed in accordance with the proposed design, as approved in the construction permit or license.

2. To determine the manner in which the reactor will be or is operated.

3. To judge the over-all safety of reactor operation, with due regard for the combined consequences of the above two determinations.

Scope of Reactor Inspection

Inspection activities generally take the following sequence:

1. Familiarization during design review and hazards evaluation activities.

2. Observation and review of construction progress.

3. Observation and review of tests of equipment and procedures before operation.

4. Observation and review of the initial charging of fuel, initial reactor start-up, and operational program.

5. Periodic observation and review during normal operation.

Reactor inspection is carried out by a small staff of reactor specialists under the assistant director for compliance of the Inspection Division, occasionally supplemented by technical people from AEC headquarters and field offices, and outside consultants chosen for their specialized knowledge. Recommendations of the Division of Inspection are forwarded to the General Manager of the AEC. Actions by the AEC with regard to the activities of licensees, when required, are taken through the Division of Licensing and Regulation.

Subjects that are considered during review and inspection of reactors include the following: status of construction; schedule and plans for, and check of, "conventional" inspections;* programs for observation and review of tests, assembly, and initial operation; physical arrangement and environment; the control system; the safety system; instrumentation; reactivity; heat generation, transfer, and distribution; kinetics and dynamics; shielding and radiation control; normal and emergency operating standards, procedures, and practices; and organization and management.

*"Conventional" inspections are those against existing engineering standards, codes, or the like, that do not involve special requirements or knowledge based on nuclear considerations.

Construction and preoperational testing of facilities have, so far, presented few problems of any direct relation to operational safety. On a number of occasions, however, detailed review of operations and equipment by inspection personnel has indicated that certain changes in design or in the method of operation could be made to improve safety. In all cases, solutions satisfactory to both licensee and the Inspection Division have been found and voluntary action has been taken by the licensee. Some examples of these situations are the following:

1. During observation of preoperational tests of a research reactor, it was found that the automatic reactor control system was so connected that shim-safety rods could be called upon to increase reactivity should the regulating rod be unable to maintain the reactor at constant power. Discussion of this situation led to changes in circuitry such that shim-safety rods cannot be moved except upon direct demand of the operator. The principle involved here is that of design of equipment such that the possibility of increase in reactivity upon malfunction is minimized.

2. In the case of a research reactor, using a homogeneous solution as fuel, it was noted that operation at a higher power planned for the future, would require the use of a fuel mixture in which the ratio of hydrogen to uranium was less than optimum. Under these conditions, the addition of water to the core would cause reactivity to increase. Such an increase could occur in the event of malfunction of certain equipment. This change in reactivity, therefore, could decrease substantially the margin of control, that is, the difference between reactivity of the core and that held by the control and safety rods. Discussion of this question prompted the licensee to reanalyze the situation in detail and to defer temporarily application to the AEC for permission to operate at the higher power.

3. Inspection of a reactor designed for research and training purposes disclosed that a number of changes to equipment could be made to enhance substantially the inherent safety of the reactor, particularly when used for training. Improvements included, for example, an interlock to provide for a minimum count rate before reactor start-up, and the removal from the control console of an easily operated control for adjusting trip levels for the safety circuits. Further consideration of training problems by the constructor led to a number of additional modifications, all of which are viewed by the constructor and the inspection group as material contributions to operational safety.

4. In the case of a completed critical facility before intial operation, it was determined that the operating staff had only very limited experience with critical experiments and reactor operation. The senior management of the organization, as a result of this observation, made effective use of a number of thoroughly experienced advisors during initial operations of the facility.

License Applications and Actions

Some of the more recent reviews of applications for construction permits or licenses to operate are discussed here. In general, only the particular applications that present unusual hazards or establish safety philosophy or precedents are discussed. The information presented in these reviews is available at the AEC Public Document Room, Washington, D. C. In addition to this source of information, reviews of all the recent license applications are given in a weekly publication, *Atomic Energy Clearing House*, published by the Congressional Information Bureau, Inc.

Yankee Atomic Electric Company Reactor

Several amendments to the Yankee Atomic Electric Company license application have been reviewed by the HEB and the ACRS and have been found to be satisfactory with the reservations noted. An amendment of interest is concerned with the effect on the reactor's nuclear parameters of the build-up of plutonium in the core over the reactor lifetime. The ACRS had originally suggested that the effect of plutonium build-up be investigated in a part-core critical assembly using synthetic fuel elements containing large-exposure plutonium. Yankee felt that such an experiment would be difficult to interpret because of the impossibility of duplicating the temperature and neutron spectrum in such a facility. Therefore Yankee proposed to conduct an experimental program to measure the temperature coefficient and other core parameters affecting operation in the actual

reactor at startup, after 2000 hr of operation, and at intervals while plutonium is building up in the core. Since independent calculations by Westinghouse Electric Corporation and Nuclear Development Corporation of America indicate that the build-up of plutonium has only a minor effect on the reactor's nuclear parameters, both the HEB and the ACRS concurred with the Yankee proposal to determine the temperature coefficients in the actual reactor. The HEB felt, however, that before an operating license was granted that Yankee should furnish a description of the specific experiments to be performed on the reactor and the experimental procedures to be followed, together with evidence that the proposed experiments and procedures would provide information necessary to predict the effect of plutonium build-up on reactor characteristics. Final action on these amendments had not been taken by the AEC as of June 1, 1959.

Dresden Nuclear Power Station

The AEC has announced that a hearing was held at AEC Headquarters on July 9, 1959, to consider the issuance of a license to the Commonwealth Edison Company to operate the Dresden plant. The report of the ACRS in this matter has indicated that this reactor can be started and operated without undue risk to the public.

The ACRS commented specifically concerning the control rods, flux monitors, and the containment vessel. The Committee felt that since the Dresden plant has a large number of control rods that depend completely on a complex hydraulic system for normal operation and for scramming that it is essential that frequent and thorough periodic checks be made of this system. Because of the large size of the reactor, both external and internal neutron flux monitors are necessary, at least during early operation, to ensure safe control of the reactor at high power levels. During early operation of the reactor at high power, it is understood that the plant will operate with at least five external ion chambers and 32 internal ion chambers.

In addition to all other safeguards, the containment shell around the reactor, which is supplemented by a well-designed emergency cooling system, will provide adequate protection for the public.

Consolidated Edison Thorium Reactor

The Consolidated Edison Company has submitted an amendment to its construction permit to cover the technical specifications for the containment vessel of the plant. Based on a review of the information contained in the application and the reports of the ACRS, some of the Commission findings are:

1. The containment vessel is designed in accordance with Section 8 of the ASME Boiler and Pressure Vessel Code (1956 edition) as modified by a number of Code cases applying to containment vessels for nuclear plants. The design is based on a working pressure of 25 psig, and the vessel will be pneumatically tested at 31.25 psig.

2. The vessel will probably have a leakage rate not in excess of 0.1 per cent per day at 25 psig.

3. Although the maximum credible accident for the proposed reactor, as presently designed, and the consequences of the accident have not been definitely established, the hypothetical accident postulated by the applicant results in consequences within acceptable limits even though the conditions are more severe than would reasonably be expected.

4. The containment vessel structure should be adequate as designed to prevent an undue hazard to the health and safety of the public, and approval was given to the mechanical design of the vessel, the internal arrangement of equipment, and the leakage analysis.

Final construction approval, however, must await further analysis of the effects of the maximum credible accident and analyses of the proposed solutions to the remaining problems:

1. A determination must be made as to whether means should be provided to relieve excessive negative internal pressure.

2. A design must be evolved for the penetrations in the sphere that will ensure the integrity of the pressure shell.

3. Adequate means must be available for closure of all penetrations that carry fluids that might be radioactive or through which the post-accident atmosphere might escape.

4. Means must be provided for rapid sealing of the sphere in the event of release of radioactivity.

5. The effect of internal missiles on the containment shell must be evaluated.

Westinghouse Testing Reactor

The Division of Licensing and Regulation of the AEC and the ACRS have reviewed the Westinghouse Testing Reactor (WTR) and have concluded that this reactor may be operated without undue hazards to the health and safety of the public.

Included in the provisions of the proposed license are several operating restrictions that are of interest in that they specify the responsibility of the applicant and of the AEC in matters of alterations to the plant.

The operating restrictions clearly establish the maximum allowable power level of the reactor, as well as the allowable excess reactivity above cold, clean critical. Furthermore, the over-all void and over-all temperature coefficients must be negative and of such values that the facility can inherently withstand, without melting of fuel cladding, a step increase in reactivity of at least 1 per cent.

The operating restrictions also provide that the operator may make the changes listed below but that these changes must be reviewed and approved by the Westinghouse Testing Reactor Safeguards Committee (WTRSC) and that the report of the group must be submitted to the AEC, who will advise Westinghouse within 15 days if approval is not granted. Included in this category of changes are:

1. Physical, nuclear, thermal, or hydraulic performance characteristics of the reactor core and/or high-pressure experimental loops or thimbles.

2. Performance characteristics of the reactor control and safety systems.

3. Number and type of experimental facilities.

4. Reactivity limitations on experiments.

5. Integrity of containment vessel.

With the exception of these restrictions, changes may be made in the facility design, performance characteristics, and operating procedures without further approval provided that an evaluation by the WTRSC of the hazards involved in the proposed change and the effect of the change on each of the postulated accidents analyzed in the final safety report do not indicate that the hazards are increased or that a material alteration of the facility is required.

In addition to these operating restrictions, the proposed license makes it necessary for the licensee to keep the following records:

1. Reactor operating records, including power level.

2. Records of activity release into the air or water beyond the effective control of Westinghouse.

3. Records of emergency shutdown and reasons therefor.

4. Records describing each change authorized by the WTRSC and a summary statement of the bases for the conclusions reached by the Committee.

5. Records containing a description of each test or experiment conducted in the facility.

Further, Westinghouse must make an immediate report in writing to the Commission on any indication or occurrence of a possibly unsafe condition relating to the operation of the facility. Westinghouse must also submit to the Commission a report on the results of operation of the facility pertinent to safety during the various power tests and identify any changes in the facility design, performance characteristics, and operating procedures. An annual report of operating experience and changes in facility design, performance characteristics, and operating procedures is also required.

A public hearing was held in March, and following this hearing the AEC hearing examiner recommended that an operating license be issued to the Westinghouse Electric Corporation.

Vallecitos Boiling Water Reactor

Many amendments have been made to the license application of the General Electric Company for the Vallecitos Boiling Water Reactor, and three of particular interest, Amendments 24, 33, and 34, are described here. Amendment 24 is designed to provide the operator with greater latitude in the choice of fuel elements and operating limits. The ACRS believes that the technical specifications set out in Amendment 24 define a scope of operation within which it is possible to operate without undue hazards to the public since the specifications do not appear to increase the magnitude of the maximum credible accident.

They feel, however, that the applicant has not demonstrated that the technical specifica-

tions will have no influence on the magnitude of the mca and that this point should be documented more definitely. They emphasize that the technical specifications alone do not guarantee safe operation of the reactor, especially from the standpoint of hazards within the boundary.

Further, they are concerned that the amendments to the licensee's application on the VBWR operation apparently stem from attempts to cover a multiplicity of specific situations pertinent to the safe operation of the reactor. Rather than improving the safety, it seems possible that the real issues of safe operation may become clouded by changing one set of circumstances for another. The ACRS suggests that the applicant be permitted within the scope of Amendment 24 to assume technical responsibility for the hazards in connection with the experimental programs. The latitude contemplated in the proposed amendment imposes a special responsibility on the applicant to review each proposed change in operating conditions in the light of its effects on the probability of an accident and on the possible severity of the accident if it should occur.

In Amendment 33, a request was made for authorization to install a nuclear superheater loop in the Vallecitos Boiling Water Reactor in which to test fuel elements. This loop will include a fuel element test facility and will consist of a high-pressure process tube located inside the reactor vessel in a fuel element position. This process tube, which is designed to superheat the primary reactor steam, will permit testing of one fuel element at a time. The fuel elements to be tested will each consist of a number of hollow, cylindrical, UO_2 pellets enclosed in concentric stainless-steel tubing to form a cylinder. The initial fuel to be tested will be enriched to about 2.3 per cent in U^{235}, and later experiments may test fuels with enrichments as high as 4.5 per cent.

The provisions of the application state that the licensee will be required to maintain a reactor scram interlock which will shut down the reactor if the steam temperature at the outlet from the loop reaches 1000°F and to observe operating procedures which will assure that the surface temperature of the cladding will not exceed 1200°F. In addition the applicant's schematic diagram for the experiments indicates that reactor scrams may be included which will be actuated by high fuel cladding temperature, low coolant flow in the exit coolant line, high coolant activity at the vent condenser, low inlet-air pressure, or low steam pressure.

The HEB feels that the scram on the steam outlet temperature, together with the responsible use of the other scrams, provides adequate protection against most of the possible accidents described by the applicant. They do not believe, however, that they ensure complete protection of the fuel elements in case of loss of coolant. On the other hand, inclusion of a scram which would shut down the reactor in the event the coolant flow through the loop was significantly diminished would provide better protection against rupture or melting of the fuel elements. Hence, it is their opinion that a scram which will shut down the reactor in the event the coolant flow through the nuclear superheater loop drops below 80 per cent of the design flow for the particular test should be provided.

It appears that no accident other than the maximum credible accident will result in a release of fission products outside of the nuclear superheater loop. Since the mca in the loop will only result in doses equal to 1/500 of those that would result from the mca of the reactor, the addition of the nuclear superheater loop to the VBWR facility and its operation in the manner described in the application, subject to the conditions described above, will not present any substantial change in the hazards evaluated previously in connection with the loop operation.

The current license authorizing operation of the General Electric Vallecitos Boiling Water Reactor restricts operation to cores containing fuel elements of which at least 75 per cent are of substantially the same design and composition. This restriction was imposed because of concern over the possibility of internal divergent power oscillations between groups of fuel elements having significantly different nuclear thermal characteristics. Reactor experiments with heterogeneous cores containing plate and rod type fuel elements have been conducted within the present license restriction and have failed to indicate the existence of such oscillations. Analytical attempts to predict the conditions required for such oscillations, or to prove

that they cannot exist with more heterogeneous cores, have been unsuccessful. Therefore, the applicant proposes in application Amendment 34 a testing and evaluation procedure to demonstrate safely whether such oscillations exist with more heterogeneous core loadings than previously authorized. A successful demonstration of safe operation will permit subsequent routine operations with core loadings essentially similar to those tested.

The applicant proposes to increase the reactor power level in a careful stepwise manner following a core loading change which might be expected to lead to internal oscillations. At each level a frequency-spectrum analysis of the noise signal from one normal out-of-core ion chamber and one or more in-core ion chambers will be made with instruments whose ranges are from 0.02 to 10 cycles/min. If the reactor demonstrates a tendency to instability, a peak will appear above the random noise spectrum at the frequency of oscillation, and an evaluation of the frequency spectrum will be made at each power level to determine whether it is safe to increase reactor power to the next level.

The HEB in general believes that the licensee's proposed procedures as modified by the license amendment will provide the proper degree of safety. However, they pointed out that from a safety point of view it is desirable to require that at least 75 per cent of any core utilized in the reactor be composed of fuel elements with characteristics that have been determined experimentally. They therefore believe that the license should provide that after it has been determined in accordance with the outlined experiment that the reactor can be safely operated with the heterogeneous core, the facility may be operated with cores composed of any combination of types of fuel elements provided at least 75 per cent of the fuel elements are of the plate type previously approved for use in the core and/or fuel elements of any other type which have successfully undergone all the tests specified in the final hazards summary report.

The Hazards Evaluation Branch also pointed out that in the application for amendment it is stated that the specified procedures will be used "following a core change which might be expected to lead to internal oscillations." This leaves the decision of whether or not to use the procedures to the licensee. Because of the present state of the knowledge and limited experience with mixed fuel loading in this reactor, there are no clear-cut criteria for making such a decision. For this reason the licensee should be required, before making any other changes in the basic core of the reactor, to conduct the following experiments, observing all the procedures of the final hazards summary report:

1. Replace approximately 12 plate type fuel elements with a like number of Dresden rod type elements.

2. If this first test is successful, the licensee should replace approximately 12 additional plate type elements with a like number of rod type elements.

Following this second change in core loading the core will consist of approximately equal numbers of plate and rod type fuel elements. Instabilities will be most likely to occur in this highly heterogeneous condition. If no instabilities are found in this condition, the HEB believes that reliance on the applicant's discretion to use the specified procedures will provide adequate protection. The licensee should be required to report to the Commission any instabilities observed during the experiments which might damage the fuel elements, and no further power increases should be permitted with the unstable loading until new procedures are approved by the Commission.

The application does not specify the location of the in-core ion chambers during the conduct of these tests; however, the HEB believes that the license should require that these chambers be placed within the boundaries of the largest clumps of grouping of similar fuel elements within the core. The ion chambers will then be in the most sensitive positions to detect instabilities. With the additions and/or changes noted above, the HEB believes that the proposed procedures described in the amendment can be executed safely and that they will not present any substantial changes in hazards and safety to the public from those already considered and evaluated in connection with the previously approved operational facility. The General Electric Company has been granted approval by the AEC to operate the Vallecitos Boiling Water Reactor as proposed.

(H. N. Culver)

Safeguards Reports and Bibliographies

It is intended that in each issue of *Nuclear Safety* in the Technical Progress Review series safeguards reports which have recently been issued will be tabulated for the reader's reference. Because of the similarity of many reactors, in particular the research reactors, the list is not intended to be all-inclusive. Bibliographies pertaining to hazards and safeguards of reactors which have recently been issued will also be included for additional reference.

Safeguards Reports

1. L. E. Stanford, Safeguard Report on the Proposed Method of Annealing Graphite in the X-10 Reactor, ORNL-2725, Apr. 20, 1959. (Unclassified AEC report.)

2. J. M. Duncan, University of Florida Training Reactor Hazards Summary Report, University of Florida Engineering and Industrial Experiment Station, October 1958.

3. Aerojet Nuclear Testing Reactor. Summary Hazards Report, AGN-3000, Aerojet-General Nucleonics, September 1958.

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5. C. G. Andre, Summary Safeguards Report for the Nuclear Test Reactor, GEAP-3068, General Electric Corp., Oct. 7, 1958.

6. Supplement to the LMFRE Preliminary Hazards Evaluation, BAW-1017-1, June 1958.

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4. H. C. Paxton et al., Nuclear Critical Accident at the Los Alamos Scientific Laboratory on December 30, 1958, LAMS-2293, February 1959. (Unclassified AEC report.)

5. USAEC Press Release No. B-30, Mar. 2, 1959.

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7. P. P. Savić, Sur l'Accident avec le Reacteur de Puissance Zero du 15 Octobre 1958, *Bulletin of the Institute of Nuclear Science, Boris Kidrich (Belgrade)*, 9: 1-4 (March 1959).

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